Progress Energy

JUN 23 2011

SERIAL: HNP-11-065
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
SUPPLEMENT TO MEASUREMENT UNCERTAINTY
RECAPTURE POWER UPRATE LICENSE AMENDEMENT REQUEST

Reference: Letter from C. Burton to the NRC, "Request for License Amendment, Measurement Uncertainty Recapture Power Uprate," dated April 28, 2011. (ADAM Accession No. ML1124A180)

Ladies and Gentlemen:

In accordance with 10 CFR 50.90, Carolina Power and Light Company (CP&L), doing business as Progress Energy Carolinas, Inc. (PEC), submitted a measurement uncertainty recapture (MUR) power uprate license amendment request (LAR) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP) that would increase the thermal rated power level from 2900 megawatts thermal (MWt) to 2948 MWt (Reference). CP&L hereby submits a supplement to the referenced LAR based on discussions held with the Nuclear Regulatory Commission (NRC) on May 26, June 7, and June 9, 2011. The supplemental information in the enclosures to this letter is provided to support NRC review of the MUR LAR.

Enclosure 2 to this letter provides updates to the "V. Electrical Equipment Design" portion of the MUR LAR, which supersede the information submitted in the referenced LAR. Updates have been made to the following sections:

1. <u>Section V.1.B - Station Blackout (SBO)</u>
   a. Section V.1.B.i - Alternate AC Power Source: This section has been corrected to clarify that HNP is an AC-independent plant and was assessed for its ability to cope with SBO for four hours without AC power.

   b. Section V.1.B.iii - Class 1E Battery Capacity: This section has been corrected to state that HNP has two Class 1E battery systems with sufficient capacity, including 10% margin, to provide power during SBO for four hours.

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P. O. Box 165
New Hill, NC 27562

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c. Section V.1.B.v - Compressed Air: This section has been corrected regarding Power Operated Relief Valve details related to SBO.

2. Section V.1.D – Grid Stability
   Section V.1.D.iii – Large Generator Interconnection Power Factor Requirements: This section has been updated to clarify the power factor requirements associated with the uprate.

3. Section V.1.F - Power Conversion Systems
   a. Section V.1.F.i - Main Generator: The section has been revised to clarify the main generator capabilities associated with the MUR Power Uprate.

   b. Section V.1.F.ii - Isolated Phase Bus: The section has been revised to explain the capability of the isolated phase bus to deliver the required MVARs relative to power factor requirements. HNP also commits to either upgrade the isolated phase bus or limit the output to the capacity of the installed isolated phase bus.

   c. Section V.1.F.iii - Main (Step-up) Transformers: This section has been updated to state the planned 425 MVA nameplate rating of the new main transformers, planned to be installed during refueling outage 17.

   d. Section V.1.F.iv – Unit Auxiliary Transformers: A typographical error was corrected in this section.

   e. Section V.1.F.v – Startup Transformers: A typographical error was corrected in this section.

Additionally, Attachment 2 to Enclosure 1 includes the retyped Operating License for the MUR Uprate. The referenced LAR submittal erroneously omitted this page.

The supplemental information provided by this letter does not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Considerations presented in the referenced LAR submittal.

This supplement contains one new Regulatory Commitment. Enclosure 3 provides the updated list of Regulatory Commitments for the MUR LAR.

Enclosures 5 and 6 contain proprietary information. Per the affidavit for withholding proprietary information (Enclosure 4), CP&L requests that the NRC withhold the information in accordance with 10 CFR 2.390. Upon removal of Enclosures 5 and 6, this letter is decontrolled.

In accordance with 10 CFR 50.91(b), HNP is providing the State of North Carolina with a copy of the proposed license amendment.

Please refer any questions regarding this submittal to Mr. Dave Corlett, Supervisor - Licensing/Regulatory Programs, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct. Executed on
[JUNE 23, 2011].

Sincerely,

William Jefferson, Jr.
Vice President
Harris Nuclear Plant

WJ/kab

Enclosures:
1. Evaluation of the Proposed Change
2. NRC Regulatory Issue Summary 2002-03 Requested Information
3. List of Regulatory Commitments
4. Affidavit for Withholding Proprietary Information
5. Cameron Engineering Report: ER-697, Revision 2 (Proprietary)
6. Cameron Engineering Report: ER-720, Revision 2 (Proprietary)

cc:
Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Mrs. B. L. Mozafari, NRC Project Manager, HNP
Mr. V. M. McCree, NRC Regional Administrator, Region II
Mr. W. L. Cox, III, N.C. DENR Section Chief

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DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
EVALUATION OF PROPOSED CHANGES

Subject: *Request for License Amendment to revise Renewed Operating License NPF-63, Technical Specifications, and associated Technical Specification Bases to reflect the uprated reactor core power level.*

ATTACHMENTS:
1. Operating License and Technical Specification Page Markups
2. Retyped Operating License and Technical Specification Pages
3. Technical Specification Bases Change (For Information Only)
4. Relocated Technical Specification and Design Basis Requirements Procedure (PLP-114), Attachment 10 (For Information Only)

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1.0 SUMMARY DESCRIPTION

This letter requests an amendment to the Renewed Facility Operating License (OL) No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit No.1 (HNP) including Appendix A, Technical Specifications (TS), to increase the licensed core power level. The proposed change will revise Renewed OL NPF-63 Maximum Power Level; Appendix A, TS definition of Rated Thermal Power (RTP); Reactor Core Safety Limits (Figure 2.1-1); Reactor Trip System Instrumentation (Table 2.2-1); Minimum Allowable Power Range Neutron Flux high setpoint with Inoperable Steam Line Safety Valves (Table 3.7-1); and TS Bases Section 3/4.7.1 to reflect the uprated reactor core power level. A new Relocated TS and Design Basis Requirements procedure (PLP-114) Attachment 10 will address the LEFM controls.

Attachment 1 provides the OL and TS change mark-ups. Attachment 2 provides the retyped OL and TS pages. Attachment 3 provides the TS Bases change (for information only). Attachment 4 provides the Relocated TS and Design Basis Requirements procedure Attachment 10 (for information only).

2.0 DETAILED DESCRIPTION

CP&L proposes to revise OL NPF-63 pursuant to 10 CFR 50.90. The measurement uncertainty recapture (MUR) power uprate License Amendment Request (LAR) increases HNP's RTP from the current 2900 megawatts thermal (MWt) to 2948 MWt, and makes TS changes as necessary to support operation at the uprated power level. The proposed change is an increase of approximately 1.66% in RTP. Unless otherwise noted, 100% power in this LAR refers to 2948 MWt. The proposed changes are shown in Table 1 and described below:

Table 1. Technical Specifications Changes

Change No.	Change Description
1	**Facility Operating License, Paragraph 2.C(1)** Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.
2	**TS 1.28, Rated Thermal Power Definition** RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2948 MWt.

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Table 1. (continued) Technical Specifications Changes

Change No.	Change Description
3	**TS Figure 2.1-1: Reactor Core Safety Limits - Three Loops in Operation with Measured RCS Flow > [293,540 GPM X(1.0 + C_1)]** Core Safety Limit lines are redrawn on this figure.
4	**TS Table 2.2-1: Reactor Trip System Instrumentation Trip Setpoints** The Total Allowance (TA) column is revised for the PR Neutron Flux - High setpoint (5.83) and Low setpoint (7.83), and the PR Neutron Flux - High Positive Rate (2.33) and High Negative Rate (2.33). **TS Table 2.2-1: Reactor Trip System Instrumentation Trip Setpoints** The PR Neutron Flux - High Trip setpoint is revised to ≤ 108%. **TS Table 2.2-1: Reactor Trip System Instrumentation Trip Setpoints** The Allowable Value column is revised for the PR Neutron Flux - High setpoint (≤ 109.5%) and PR Neutron Flux - Low setpoint (≤ 26.8%).
5	**TS Table 3.7-1: Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 3 Loop Operation** The revised values are 49%, 32%, and 15% of RTP for one, two and three inoperable main steam safety valves respectively.

2.1 Facility Operating License, Paragraph 2.C(1)
TS 1.28, Rated Thermal Power Definition

The HNP OL (NPF-63) specifies the maximum steady state core thermal power level for plant operation. The current authorized power level is 2900 MWt. The MUR power uprate will increase power by approximately 1.66%. This increase is based on a plant specific evaluation of reactor power measurement uncertainty using the LEFM instrumentation versus the 10 CFR 50, Appendix K previously mandated 2.0% uncertainty. Therefore, the new RTP will be:

RTP = 2900 MWt * 1.01664 (2.0% - 0.336%) = 2948 MWt (rounded down for conservatism)

Detailed evaluations and analyses were performed demonstrating that HNP operation at a reactor power level of 2948 MWt is acceptable. The detailed evaluations and analyses considered the effects of operation at this power level on: power level measurement uncertainty; postulated accidents and transients; mechanical, structural and material components integrity and design; electrical equipment design; system design; operator actions, emergency and abnormal operating procedures, control room, plant simulator, and operator training; and environmental impact. The evaluations and analyses were performed with the standard NRC approved core models and methodologies used in the existing core design and safety analyses. Normal operating and accident conditions were evaluated at the uprate conditions to confirm that safety limits and design criteria were met. These evaluations and analyses are described in Enclosure 2.

2.2 TS Figure 2.1-1: Reactor Core Safety Limits - Three Loops in Operation with Measured RCS Flow > [293,540 GPM X (1.0 + C_1)]

The core safety limit lines are redrawn as indicated on the TS retyped pages (Attachment 2). The power uprate impact on the core thermal hydraulic design was evaluated for Final Safety Analysis Report (FSAR) Chapter 15 transients with regard to DNB and core safety limits. A statistical evaluation of the core safety limit lines using the currently approved methods and codes took into account the higher nominal power, reduced uncertainty, and design axial shapes at uprate conditions. The evaluation concluded that the existing core safety limit lines should be modified for consistency with the plant setpoints evaluation. The revised core safety limit lines data points are shown in Table 2 below.

Table 2. Data Points for Redrawn Core Safety Limit Lines

Core Safety Limit Lines Data Points					
2385 psig		2235 psig		1960 psig	
Power Fraction	Temperature (°F)	Power Fraction	Temperature (°F)	Power Fraction	Temperature (°F)
0.00	654.75	0.00	645.25	0.00	627.00
0.96	625.00	0.96	613.00	0.96	598.00
1.20	599.00	1.20	589.00	1.20	575.00

2.3 TS Table 2.2-1: Reactor Trip System Instrumentation Trip Setpoints

TS Table 2.2-1 describes the reactor protection system functions. HNP employs a five-column format for this table. The columns are: Total Allowance (TA), Z - a factor that accounts for statistical summation of errors, Sensor Error (S), Trip Setpoint, and Allowable Value. Changes are proposed for the following reactor trips:

- Power Range, Neutron Flux High Setpoint
- Power Range, Neutron Flux Low Setpoint
- Power Range, Neutron Flux, High Positive Rate
- Power Range, Neutron Flux, High Negative Rate

The TA values for the four reactor trips listed above require revision. The TA term represents the difference between the safety analysis limit (SAL) and the nominal trip setpoint.

The safety analysis trip setpoints in terms of absolute power are unchanged for the Power Range Neutron Flux Low, Power Range Neutron Flux High Positive Rate and Power Range Neutron Flux High Negative Rate reactor trips. However, these trips are listed in terms of percent RTP; therefore the safety analysis trip setpoints are re-designated based on the ratio of the current RTP to the uprated RTP (2900/2948). The TA for these three trips is recalculated to reflect the 1.66% increase in RTP.

The Power Range, Neutron Flux Low Setpoint Allowable Value is reduced from 27.1% of RTP to 26.8% of RTP. The current calculation of record (Reference 6.4) Allowable Value is 26.8% of RTP, which resulted from the HNP SG replacement/power uprate in 2001. However, the original 27.1% Allowable Value was retained at that time to preclude a TS change, since it was conservative. The MUR power uprate is changing the Power Range Neutron Flux - Low Setpoint Allowable Value to match the current calculation of record.

The Power Range, Neutron Flux High Setpoint has additional impact beyond the TA change. The evaluation of non-LOCA events at power uprate conditions concluded that the results are acceptable and no trip setpoint changes were required. However, the margins for events that use statistical DNB methodology are fractionally affected in the negative (non conservative) direction by the reduction in power uncertainty from 2.0% to 0.34%. Operating at a known slightly higher power level with less uncertainty resulted in a minimum DNBR with less margin to the safety analysis limit for the limiting event (uncontrolled rod withdrawal at power). To retain design margin for future cycle variability, the SAL for the Power Range, Neutron Flux

High was reduced from 118% to 117% in terms of the current RTP. Considering the uprated power level, the 117% SAL was further reduced to 115% (117% x (2900/2948)). This change increases the DNB margin for limiting events to satisfy HNP objectives for design margin. The SAL change impacts the Power Range Neutron Flux High TA, Trip Setpoint, and Allowable Value.

Table 3. TS Table 2.2-1 Reactor Trip System Instrumentation Trip Setpoint Changes

		Current Value	**MUR Value**
Power Range Neutron Flux - High Setpoint	Total Allowance	7.5	5.83
	Trip Setpoint	≤ 109% of RTP	≤ 108% of RTP
	Allowable Value	≤ 111.1% of RTP	≤ 109.5% of RTP
Power Range Neutron Flux - Low Setpoint	Total Allowance	8.3	7.83
	Allowable Value	≤ 27.1% of RTP	≤ 26.8% of RTP
Power Range Neutron Flux - High Positive Rate	Total Allowance	2.5	2.33
Power Range Neutron Flux - High Negative Rate	Total Allowance	2.5	2.33

The following two notes are added to TS Table 2.2-1 Functional Units 2.a (Power Range Neutron Flux - High Setpoint), 2.b (Power Range Neutron Flux - Low Setpoint), 3 (Power Range Neutron Flux - High Positive Rate) and 4 (Power Range Neutron Flux - High Negative Rate).

Note 7 If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

Note 8 The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. NTSPs more conservative than the Trip Setpoints in Table 2.2.1 are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in Progress Energy procedure EGR-NGGC-0153, "Engineering Instrument Setpoints."

2.4 TS Table 3.7-1: Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 3 Loop Operation

If one or more main steam safety valves (MSSV) are inoperable and not restored within four hours, the Power Range Neutron Flux High trip setpoint is reduced based on the number of inoperable safety valves on any operating steam generator. The maximum allowable Power Range Neutron Flux High trip setpoint is the maximum power level corresponding to the heat removal capability of the remaining operable MSSVs. The maximum allowable power levels were calculated based on the nominal rated power (2900 MWt). To maintain the same absolute power in terms of MWt as the current analysis of record, the maximum allowable Power Range Neutron Flux High trip setpoints in TS Table 3.7-1 are reduced to 49%, 32%, and 15% of uprated RTP for one, two and three inoperable MSSVs on any operating SG respectively.

2.5 TS Bases B 3/4.7.1 and Relocated TS and Design Basis Requirements Procedure (PLP-114)

TS Bases and PLP-114 changes are being made to support this LAR. The specific changes are TS Bases Section B 3/4.7.1 (Main Steam Safety Valves) and a new Attachment 10 to PLP-114 for the LEFM controls. The TS Bases and PLP-114 changes are provided for information only in Attachments 3 and 4 respectively. The PLP-114 procedure is incorporated by reference in FSAR Section 13.5.1. As stated in FSAR Section 13.5.1, PLP-114 changes are controlled using the 10 CFR 50.59 process.

2.6 Final Safety Analysis Report

Changes to the FSAR are being made to support this LAR. These changes will be made in accordance with 10 CFR 50.59.

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3.0 TECHNICAL EVALUATION

HNP is presently licensed for a RTP of 2900 MWt. More accurate feedwater flow measurement equipment results in a maximum core thermal power measurement uncertainty of 0.336%. Based on the reduction in thermal power uncertainty, the ultrasonic flow measuring equipment supports a RTP increase up to 1.664% (i.e., 2.0% - 0.336%). This increase corresponds to 2948.256 MWt, which is rounded down to the requested 2948 MWt or approximately 1.66% increase. The power uprate evaluations addressed the following: nuclear steam supply system (NSSS) performance parameters, accidents, design transients, systems, components, nuclear fuel, and interfaces between NSSS and balance-of-plant (BOP) systems. The evaluation conclusions are summarized in Enclosure 2, information requested by NRC Regulatory Issue Summary (RIS) 2002-03. These analyses were reviewed to provide assurances that they remain bounding for the proposed power uprate. There are no non-bounding analyses.

The NSSS design thermal and hydraulic parameters derived from the power uprate conditions serve as the basis for the NSSS analyses. A detailed review of the accident analyses was performed for the steam generator tube rupture, loss of coolant accident (LOCA), and non-LOCA events. The currently assumed mass and energy releases remain bounding. The accident doses at power uprate conditions do not exceed the applicable dose acceptance. The fuel was evaluated for its ability to perform at the uprated power level. CP&L concludes that the changes to the HNP design basis and transient analyses are acceptable. Each of the NSSS systems and components were evaluated at the uprated conditions. The BOP systems, electrical power systems, control systems and instrumentation systems were also evaluated at uprated conditions. The analyses and evaluations performed demonstrate that the acceptance criteria continue to be met. HNP intends to complete the plant modifications needed to support operation at the uprated conditions prior to power uprate implementation, as stated in Enclosure 2, Section VII.3.

3.1 Nuclear Steam Supply System Design Parameters

The NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. The design parameters are established using conservative input assumptions to provide bounding conditions used in the NSSS analyses. They provide the primary and secondary side system conditions (thermal power, temperatures, pressures, flows) that are used as the basis for the NSSS analyses and evaluations. The NSSS design parameters were revised as shown in Table 4. These parameters have been incorporated, as required, into the applicable NSSS system and component evaluations, and safety analyses performed to support the power uprate.

3.1.1 Input Parameters

The major input parameters used to calculate the four cases of NSSS design parameters are as follows:

- Bounding reactor core power level of 2958 MWt (NSSS power of 2970.4 MWt, 102% of 2900 MWt + 12.4 MWt RCP heat input)
- Two core bypass flows were used: 8.6%, which accounts for thimble plugs removed, and 7.1%, which accounts for thimble plugs installed
- AREVA 17x17 fuel
- Feedwater temperature range of 375.0°F to 440.0°F
- Westinghouse Delta-75 model replacement steam generators (SG)
- Vessel average temperature (T_{avg}) range of 572°F to 588.8°F
- Steam generator tube plugging (SGTP) levels of 0% and 10%
- Thermal design flow of 92,600 gpm/loop
- Reactor coolant pressure of 2250 psia, which is the current operating value

3.1.2 Parameter Cases

The computer code used to determine the NSSS design parameters was NSSSPlus. There is no explicit NRC approval for this code because it is used to facilitate calculations that could be performed by hand. Four cases of NSSS design parameters were used as the basis for the NSSS analytical evaluations. The appropriate design parameters were used for each NSSS analysis, based on the conditions that were most limiting for that analytical area. These four cases are shown in Table 4.

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Table 4. Revised NSSS Design Parameters for Harris Nuclear Plant MUR Uprating

Parameter	Current Design Conditions	Case 1	Case 2	Case 3	Case 4
THERMAL DESIGN					
NSSS Power, %	100	102	102	102	102
MWt	2912.4	2970.4	2970.4	2970.4	2970.4
10^6 BTU/hr	9938	10,135	10,135	10,135	10,135
Reactor Power, MWt	2900	2958	2958	2958	2958
10^6 BTU/hr	9895	10,093	10,093	10,093	10,093
Thermal Design Flow, gpm/loop	92,600	92,600	92,600	92,600	92,600
Reactor 10^6 lb/hr	103.8	106.4	106.4	103.9	103.9
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	7.1	8.6(1,2)	8.6(1,2)	8.6(1,3)	8.6(1,3)
Reactor Coolant Temperature, °F					
Core Outlet	627.7	614.0(2)	614.0(2)	629.4(3)	629.4(3)
Vessel Outlet	623.2	608.0	608.0	623.8	623.8
Core Average	593.4	577.1(2)	577.1(2)	594.2(3)	594.2(3)
Vessel Average	588.8	572.0	572.0	588.8	588.8
Vessel/Core Inlet	554.4	536.0	536.0	553.8	553.8
Steam Generator Outlet	554.1	535.7	535.7	553.5	553.5
Steam Generator					
Steam Temperature, °F	539.9	522.4	520.3	540.7(4)	538.6
Steam Pressure, psia	962	830	815	968(4)	952
Steam Flow, 10^6 lb/hr total	11.77/12.84	11.94/13.01	11.93/13.01	12.01/13.10(4)	12.00/13.09
Feedwater Temperature, °F	375.0/440	375.0/440.0	375.0/440.0	375.0/440.0	375.0/440.0
Steam Moisture, % max.	0.10	0.10	0.10	0.10	0.10
Tube Plugging, %	10	0	10	0	10
Zero Load Temperature, °F	557	557	557	557	557
HYDRAULIC DESIGN					
Mechanical Design Flow, gpm/loop	107,100	107,100	107,100	107,100	107,100

Notes:
1. Core bypass flow accounts for thimble plugs removed.
2. If thimble plugs are installed, the core bypass flow is 7.1%, core outlet temperature is 612.9°F, and core average temperature is 576.5°F.
3. If thimble plugs are installed, the core bypass flow is 7.1%, core outlet temperature is 628.4°F, and core average temperature is 593.6°F.
4. For analyses limited by high steam pressure, conditions corresponding to a maximum steam pressure of 997 psia, steam temperature of 544.2°F, and total steam flow of 13.12 x 10^6 lb/hr are assumed. This covers the possibility that the plant could operate with better than expected steam generator performance.

Cases 1 and 2 represent parameters based on a T_{avg} of 572.0°F. Case 2 is based on an average 10% SGTP and yields the minimum SG secondary side steam pressure and temperature. All primary side temperatures are identical for these two cases.

Cases 3 and 4 represent parameters based on a T_{avg} of 588.8°F. Case 3 is based on an average 0% SGTP and yields the maximum SG secondary side steam pressure and temperature. All primary side temperatures are identical these two cases. The data provided in Note 4 of Table 4 was used in those NSSS analyses and evaluations that require an absolute upper limit steam pressure. This more limiting secondary side data are based on the Case 3 parameters, with an assumed SG fouling factor of zero.

3.2 Best Estimate Performance Parameters

Best estimate performance predictions were calculated for the steam conditions at the SG outlet, as opposed to the conservative design conditions shown in Table 4. The best estimate performance predictions were based on plant calorimetric data from HNP Cycle 15. The major input parameters and assumptions used to calculate the three cases of best estimate performance predictions are as follows:

- NSSS power of 2961.7 MWt (101.7% of 2900 MWt + 12.4 MWt RCP heat input)
- Westinghouse Delta-75 model replacement SG
- Total SG blowdown flow of 106,215 lbm/hr
- Best estimate fouling factor based on plant calorimetric data
- Moisture carryover of 0.10%

The computer code used to determine the best estimate performance predictions was NSSSPlus. There is no explicit NRC approval for this code because it is used to facilitate calculations that could be performed by hand. The results of the three cases are shown in Table 5. The best estimate performance prediction results were used, as appropriate, for evaluating the MUR power uprate effects on BOP systems.

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Table 5. Calorimetric Data-Based Best Estimate Performance Predictions

NSSS Power (MWt)	SGTP (%)	T_{inlet} (°F)	T_{avg} (°F)	T_{feed} (°F)	RCS Flow [1] (gpm/loop)	Steam Pressure [2] (psia)	Total Steam Flow (lbm/hr)
2961.7	0.037	556.9	588.8	434.7	101,310	1004	12.960E6
2961.7	0.037	554.6	586.7	434.7	101,310	985	12.951E6
2961.7	10	556.2	588.8	434.7	99,089	985	12.951E6

Notes:
1. RCS flows were calculated based on plant calorimetric data.
2. Steam pressure at the SG nozzle outlet, just downstream of the flow restrictor.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

HNP was initially licensed to operate at a maximum of 2775 MWt. In Amendment 107, dated October 12, 2001, the NRC approved HNP operation at the current power level of 2900 MWt. The proposed MUR power uprate is based on a redistribution of analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed per the requirements of 10 CFR 50, Appendix K, *ECCS Evaluation Models*. Appendix K originally required 102% of licensed power level for light water reactor ECCS evaluation models. The NRC approved a change to the 10 CFR 50, Appendix K requirements on June 1, 2000 effective July 31, 2000. This change provided licensees the option of maintaining the 2.0% power margin between licensed power level and the ECCS evaluation assumed power level, or applying a reduced ECCS evaluation margin based on an accounting of uncertainties due to instrumentation error.

Implementing the Cameron LEFM CheckPlus System is an effective way to obtain additional plant power without significantly changing current reactor core operations. Feedwater flow measurement uncertainty is the most significant contributor to core power measurement uncertainty. The LEFM provides a more accurate measurement of feedwater flow and thus reduces the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty of 0.34% at RTP.

The LEFM provides on-line main feedwater flow and temperature measurement to determine reactor thermal power. This system uses acoustic energy pulses to determine the main feedwater mass flow rate and temperature. The LEFM consists of a measuring section containing 16 ultrasonic multi-path transit time transducers divided into two planes of eight, one dual resistance temperature detector (RTD) and two pressure transmitters installed in each of the three feedwater lines, and an electronic signal processing cabinet.

The LEFM will be used in lieu of the current venturi-based feedwater flow indication and RTD temperature indication to perform the plant calorimetric measurement calculation. The currently installed venturi-based feedwater flow instruments will continue to provide inputs to other indication, protection and control systems, and will be used if the LEFM is inoperable.

4.2 Precedent

License amendment applications based on the Cameron LEFM CheckPlus System were previously approved for PWRs Calvert Cliffs 1 & 2 (Reference 6.1), North Anna 1 & 2 (Reference 6.2) and Surry 1 & 2 (Reference 6.3). These submittals requested NRC approval to increase licensed power level by reducing uncertainty through the use of the LEFM CheckPlus System for feedwater flow measurement. The HNP submittal is comparable to those license amendment applications.

4.3 Significant Hazards Consideration

Carolina Power and Light Company (CP&L), doing business as Progress Energy Carolinas, Inc. (PEC), has evaluated this License Amendment Request (LAR) against the 10 CFR 50.92 criteria to determine if any significant hazards consideration is involved, and concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92(c) criteria is satisfied.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO.1
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1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change will increase the Shearon Harris Nuclear Power Plant, Unit No.1 (HNP) rated thermal power (RTP) from 2900 megawatts thermal (MWt) to 2948 MWt. Nuclear steam supply system and balance-of-plant systems, components and analyses that could be affected by the proposed change in RTP were evaluated using revised design parameters. The evaluations determined that these structures, systems and components are capable of performing their design function at the proposed uprated RTP of 2948 MWt. An evaluation of the accident analyses demonstrated that the applicable analysis acceptance criteria are still met with the proposed changes. Power level is an input assumption to equipment design and accident analyses, but it is not a transient or accident initiator. Accident initiators are not affected by the power uprate, and plant safety barrier challenges are not created by the proposed changes.

The radiological consequences of operation at the uprated power conditions have been assessed. The proposed change in RTP does not affect release paths, frequency of release, or the analyzed source term for any accidents previously evaluated in the HNP Final Safety Analysis Report. Structures, systems and components required to mitigate transients are capable of performing their design functions with the proposed changes, and are thus acceptable. Analyses performed to assess the effects of mass and energy releases remain valid. The source term used to assess radiological consequences was reviewed and determined to bound operation at the proposed power level.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of any proposed changes. The ultrasonic flow meter has been analyzed, and system failures will not adversely affect any safety-related system or any structures, systems or components required for transient mitigation. Structures, systems and components previously required for transient mitigation are still capable of fulfilling their intended design functions. The proposed changes have no significant adverse affect on any safety-related structures, systems or components and do not significantly change the performance or integrity of any safety-related system.

The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Plant operation at a RTP of 2948 MWt does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed changes.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, reactor coolant pressure boundary, and containment barriers. Core analyses demonstrate that power uprate implementation will continue to meet the current nuclear design basis. Impacts to components associated with the reactor coolant system pressure boundary structural integrity, and factors such as pressure-temperature limits, vessel fluence, and pressurized thermal shock were evaluated and determined to be acceptable at the uprate power level.

Systems will continue to operate within their design parameters and remain capable of performing their intended safety functions following power uprate implementation. The current HNP safety analyses, including the design basis radiological accident dose calculations, bound the power uprate.

Therefore, this change does not involve a significant reduction in a margin of safety.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the public health and safety will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to public health and safety.

Based on the above, CP&L concludes that the proposed license amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed facility operating license amendment requires no environmental assessment if facility operation per the proposed amendment would not: (i) involve a significant hazards consideration, (ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure.

CP&L has determined that this license amendment request meets the criteria for categorical exclusion set forth in 10 CFR 51.22(b). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment is required in connection with issuance of the proposed license amendment. This determination is based on the following:

(i) The license amendment request does not involve a significant hazards consideration, as described in the significant hazards evaluation.

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(ii) The proposed change does not involve new equipment or modifying any existing equipment that might affect the types or amounts of effluents released offsite

There will be no significant change in the types or significant increase in the amounts of any effluents released offsite during normal operation. The primary coolant specific activity is expected to increase by no more than the percentage increase in power level. Gaseous and liquid radwaste effluent activity is expected to increase from current levels by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be within allowable 10 CFR 20 and 10 CFR 50, Appendix I limits per the HNP Offsite Dose Calculation Manual. The proposed changes will not result in changes to the operation or design of the gaseous or liquid waste systems and will not create any new or different radiological release pathways.

Solid radwaste effluent activity is expected to increase from current levels proportionally to the increase in long half-life coolant activity. The total long-lived activity is expected to be bounded by the percent of power uprate. Changes in solid waste volume are expected to be minor.

During power uprate operation, the non-radiological effluents will continue to be within regulatory standards for offsite outfalls and internal onsite outfalls.

Therefore, the proposed license amendment request will not result in a significant change in the types or significant increase in the amounts of effluents that may be released offsite.

(iii) The license amendment request does not significantly increase core power and resultant dose rates in accessible plant areas. Normal operation radiation levels will increase by approximately the percentage of core power uprate. The power uprate does not require additional radiation shielding to support normal plant operation. Individual worker exposures will be maintained within regulatory limits and ALARA by the HNP Health Physics Program, which controls access to radiation areas and maintains compliance with 10 CFR 20.

Therefore, the license amendment request does not result in a significant increase to the individual or cumulative occupational radiation exposure.

6.0 REFERENCES

6.1 Letter from Douglas V. Pickett (USNRC) to James A. Spina (Calvert Cliffs Nuclear Power Plant Inc.), *Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment RE: Measurement Uncertainty Recapture Power Uprate,* ML091820366, July 22, 2009.

6.2 Letter from V. Sreenivas (USNRC) to David A. Heacock (Virginia Electric and Power Company), *North Anna Power Station, Unit Nos. 1 and 2 - Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate,* ML092250616, October 22, 2009.

6.3 Letter from Karen Cotton (USNRC) to David A. Heacock (Virginia Electric and Power Company), *Surry Power Station; Unit Nos. 1 and 2 - Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate,* ML101750002, September 24, 2010.

6.4 HNP Calculation HNP-I/INST-1010, Revision 4, *Evaluation of Tech Spec Related Setpoints, Allowable Values, and Uncertainties Associated with RTS/ESFAS Functions for Steam Generator Replacement (with Current 2787 MWT-NSSS Power or Uprate to 2912.4 MWT-NSSS Power),* February 21, 2011.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO.1
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ATTACHMENT 1
OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGE MARKUPS
(7 Pages)

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

2948

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of ~~2900~~ megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

DELETE

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. ~~134~~, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at §15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

[1]The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

DELETE

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

2948

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2900~~ MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

DELETE

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.31 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.32 For these Specifications, the SITE BOUNDARY shall be identical to the EXCLUSION AREA BOUNDARY defined above.

DELETE

Replace with
INSERT A

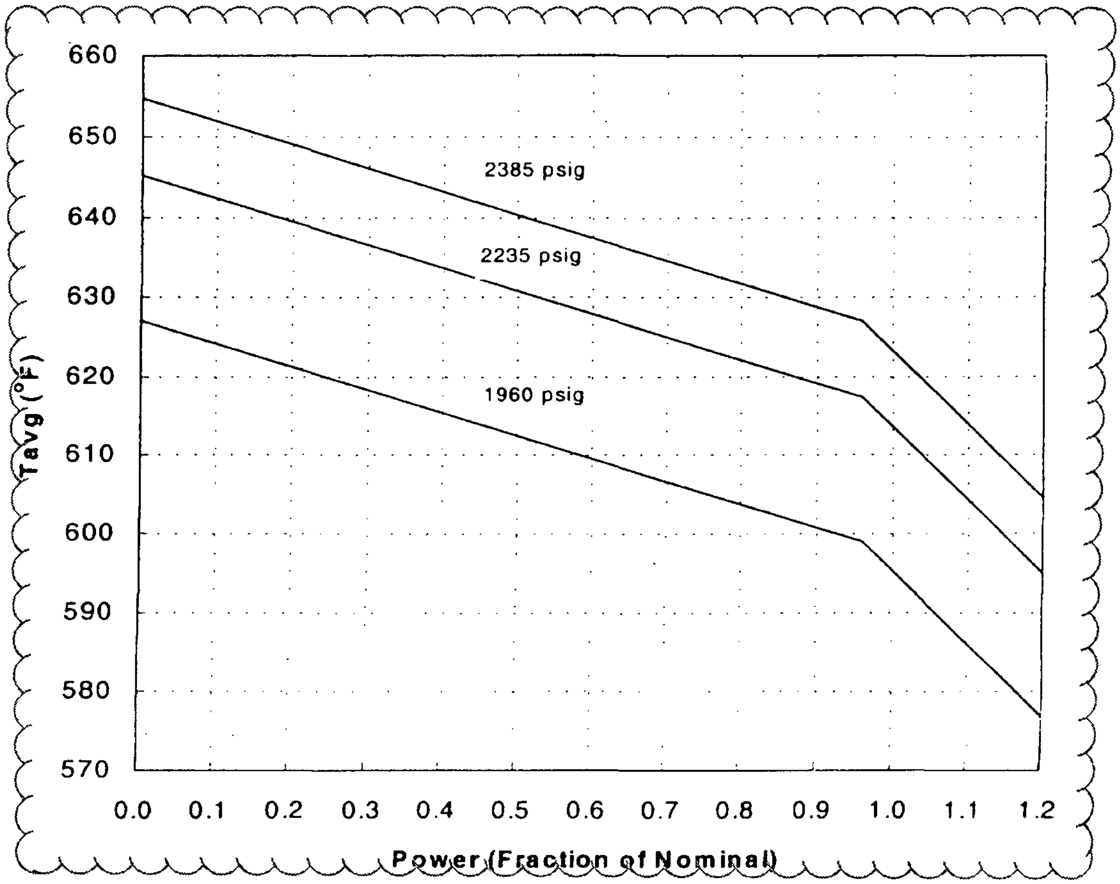


FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION
WITH MEASURED RCS FLOW > [293,540 GPM X (1.0 + C_1)]

DELETE

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	~~7.5~~ 5.83	4.56	0	≤ ~~109~~ 108% of RTP**	≤ ~~111.1~~ 109.5% of RTP** INSERT B
b. Low Setpoint	~~8.3~~ 7.83	4.56	0	≤ 25% of RTP**	≤ ~~27.1~~ 26.8% of RTP** INSERT B
3. Power Range, Neutron Flux, High Positive Rate	~~2.5~~ 2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds INSERT B
4. Power Range, Neutron Flux, High Negative Rate	~~2.5~~ 2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds INSERT B
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT	9.0	7.31	Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10. Pressurizer Pressure-High	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11. Pressurizer Water Level-High	8.0	3.42	1.75	≤ 92% of instrument span	≤ 93.5% of instrument span

DELETE

**RTP = RATED THERMAL POWER

DELETE

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_6	=	0.002/°F for T > T" and K_6 = 0 for T ≤ T",
T	=	As defined in Note 1.
T"	=	Reference T_{avg} at RATED THERMAL POWER (≤588.8°F).
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

DELETE

DELETE

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

INSERT C

DELETE

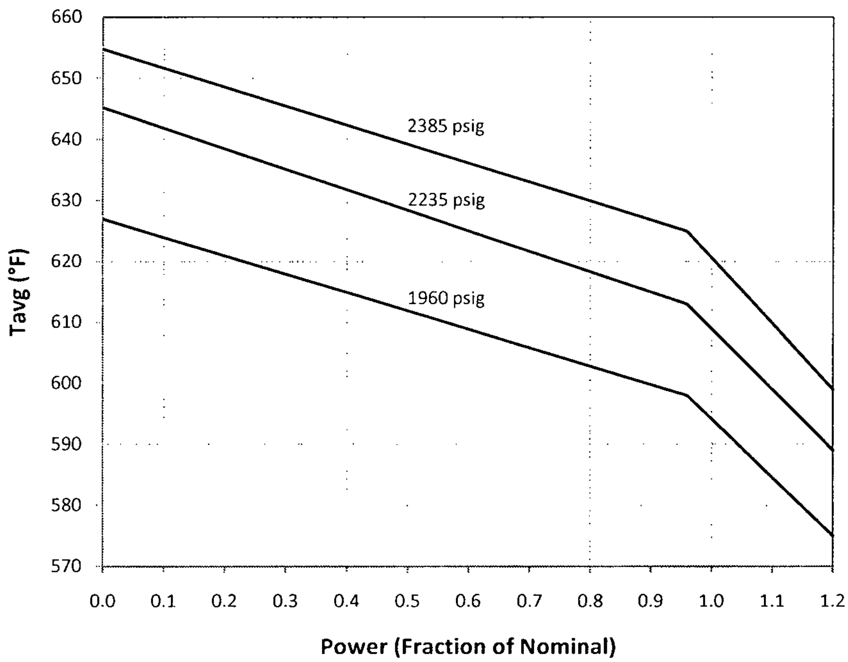
TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	~~50~~ 49
2	~~33~~ 32
3	~~16~~ 15

DELETE

INSERT A:



INSERT B:

SEE NOTES 7, 8

INSERT C:

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. NTSPs more conservative than the Trip Setpoints in Table 2.2-1 are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in Progress Energy Procedure EGR-NGGC-0153, "Engineering Instrument Setpoints."

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO.1
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REQUEST FOR LICENSE AMENDMENT
EVALUATION OF PROPOSED CHANGES

ATTACHMENT 2
RETYPED OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGES
(6 Pages)

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

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The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

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Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at §15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

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1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

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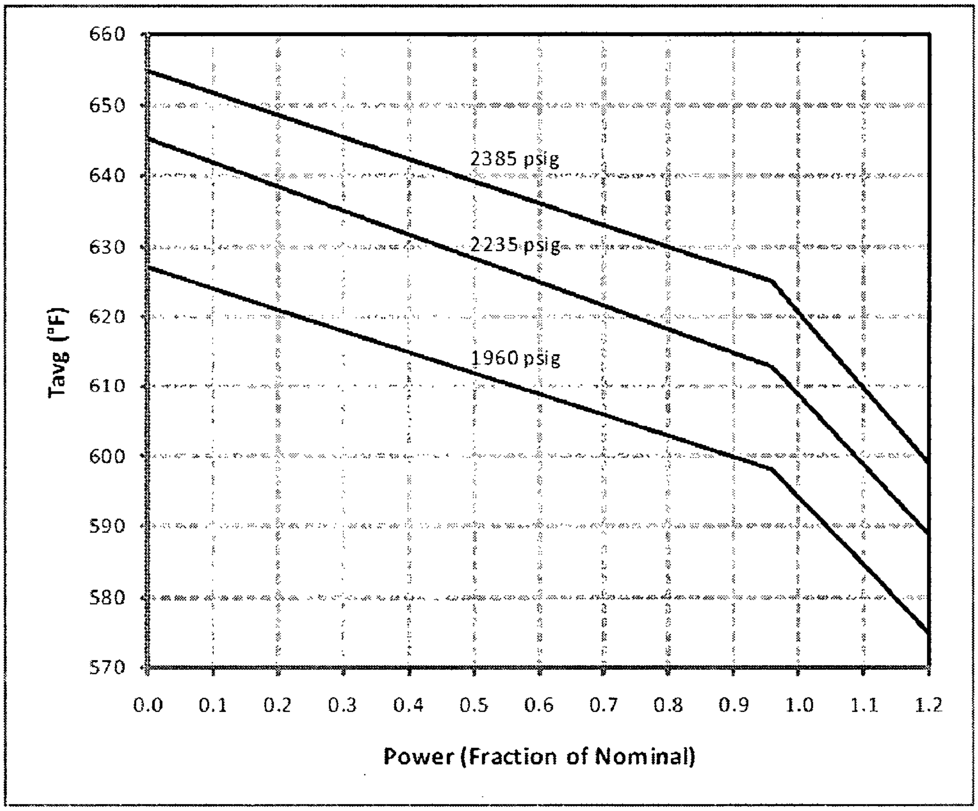

FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION
WITH MEASURED RCS FLOW ≥ [293,540 GPM X (1.0 + C_1)]

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a. High Setpoint	5.83	4.56	0	≤ 108% of RTP**	≤ 109.5% of RTP** See NOTES 7, 8
b. Low Setpoint	7.83	4.56	0	≤ 25% of RTP**	≤ 26.8% of RTP** See NOTES 7, 8
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4. Power Range, Neutron Flux, High Negative Rate	2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds, See NOTES 7,8
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
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9. Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
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**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

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NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.

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NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that is is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Normal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. NTSPs more conservative than the Trip Setpoints in Table 2.2-2 are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSP and the as-found and the as-left tolerances are specified in Progress Energy Procedure EGR-NGGC-0153, "Engineering Instrument Setpoints."

TABLE 3.7-1

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1	49
2	32
3	15

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO.1
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ATTACHMENT 3
TECHNICAL SPECIFICATION BASES MARKUP AND RETYPED PAGES
(For Information Only)
(2 Pages)

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

1.39

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~1.36~~ x 10^7 lbs/h which is in excess of 105% of the maximum calculated steam flow of ~~12.9~~ x 10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving
13.12 capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For 3 loop operation

$$Hi\varnothing = (100/Q)\frac{(w_s\ h_{fg}\ N)}{K}$$

Where:

$Hi\varnothing$ = Safety Analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{(\text{Btu/sec})}{\text{Mwt}}$

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

The values from this algorithm must then be adjusted lower to account for instrument and channel uncertainties. This adjustment will be 9% power.

DELETE

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.39×10^7 lbs/h which is in excess of 105% of the maximum calculated steam flow of 13.12×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For 3 loop operation

$$Hi\varnothing = (100/Q)\frac{(w_s\, h_{fg}\, N)}{K}$$

Where:

$Hi\varnothing$ = Safety Analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{(Btu/sec)}{Mwt}$

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

The values from this algorithm must then be adjusted lower to account for instrument and channel uncertainties. This adjustment will be 9% power.

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EVALUATION OF PROPOSED CHANGES

ATTACHMENT 4
TECHNICAL SPECIFICATION AND DESIGN BASIS REQUIREMENTS
PROCEDURE (PLP-114) ATTACHMENT 10
(For Information Only)
(2 Pages)

Feedwater LEFM Calorimetric

1.0 OPERATIONAL REQUIREMENTS

1.1 The Feedwater LEFM Calorimetric shall be OPERABLE with:

a. The Feedwater LEFM System in the Normal mode.

b. The ERFIS calorimetric program OPERABLE.

APPLICABILITY: MODE 1 with THERMAL POWER > 2900 MWt (98.4% RTP)

ACTION:

a. With the Feedwater LEFM System in the Fail mode, change the calorimetric program from the Feedwater LEFM System to the Normalized Feedwater Venturi System within 1 hour and either:

(1) Restore the Feedwater LEFM System to the Normal mode within 72 hours, or reduce THERMAL POWER to ≤ 2900 MWt (98.4% RTP) and change the calorimetric program from the Normalized Feedwater Venturi System to the Feedwater Venturi System. If the plant experiences a power decrease below 2900 MWt (98.4% of RTP) during the 72 hour allowed outage time, the maximum permitted power level will be 2900 MWt until the LEFM is restored to either Normal or Maintenance mode operation.

OR

(2) Restore the Feedwater LEFM System to the Maintenance mode within 72 hours and reduce THERMAL POWER to ≤ 2943 MWt (99.86% RTP). The plant can operate at this power level indefinitely.

b. With the Feedwater LEFM System in the Maintenance mode, restore the Feedwater LEFM System to the Normal mode within 72 hours, or reduce THERMAL POWER to ≤ 2943 MWt (99.86% RTP). The plant can operate at this power level indefinitely.

c. With the ERFIS calorimetric program inoperable for reasons other than the LEFM System, restore the ERFIS calorimetric program to OPERABLE status prior to performing the next required power range channel calorimetric heat balance comparison per TS Table 4.3-1, channel calibration D2, or reduce THERMAL POWER to ≤ 2900 MWt (98.4% RTP) by monitoring alternate power indications.

2.0 <u>SURVEILLANCE REQUIREMENTS</u>

2.1 Perform the power range channel calorimetric heat balance comparison per TS Table 4.3-1, channel calibration D2 using the Feedwater LEFM System prior to exceeding 2900 MWt (98.4% RTP) and once per 24 hours thereafter.

2.2 Perform Channel Calibration of the Feedwater LEFM System instrumentation once per 18 months.

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Acronym List

Expression	Definition or Use
AC	alternating current
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
AMSAC	ATWS mitigation system actuation circuitry
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOR	analysis of record
AOV	air operated valve
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
AST	alternative source term
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BOP	balance of plant
B&PV	boiler and pressure vessel
BWR	boiling water reactor
CCW	component cooling water
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CP&L	Carolina Power and Light Company
CPU	central processing unit
CRDM	control rod drive mechanism
CS	containment spray
CST	condensate storage tank
CUF	cumulative usage factors
CVCS	chemical and volume control system
DBA	design basis accident
DC	direct current
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDG	emergency diesel generator

Acronym List (continued)

EFPY	effective full power years
EOL	end of life
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EQ	environmental qualification (10 CFR 50.49)
ERFIS	emergency response facility information system
ESF	engineered safety features
ESFAS	engineered safety features actuation system
ESS	extraction steam system
ESW	emergency service water
FAC	flow accelerated corrosion
FHB	fuel handling building
FSAR	Final Safety Analysis Report
FW	feedwater
GL	Generic Letter
gpm	gallons per minute
HELB	high energy line break
HFP	hot full power
HNP	Harris Nuclear Plant
hp	horsepower
HTP	high temperature performance
HZP	hot zero power
IEEE	Institute of Electrical and Electronics Engineers
ISI	inservice inspection
IST	inservice testing
kW	kilowatt
LAR	license amendment request
LBLOCA	large break loss of coolant accident
LCO	limiting condition for operation
LEFM	leading edge flow meter
LOCA	loss of coolant accident
LOOP	loss of offsite power
LPZ	low population zone
LTOP	low temperature overpressure protection

Acronym List (continued)

MELB	moderate energy line break
MeV	mega electron volts
MOV	motor operated valve
MSIV	main steam isolation valve
MSLB	main steam line break
MSR	moisture separator reheater
MSS	main steam system
MSSV	main steam safety valve
MUR	measurement uncertainty recapture
MVA	mega volt ampere
MVAR	mega volt ampere reactive
MWD/MTU	mega watt day per metric ton uranium
MWe	mega watt electric
MWt	mega watt thermal
NERC	North American Electric Reliability Corporation
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PEC	Progress Energy Carolinas, Inc.
PORV	power operated relief valve
psi	pounds per square inch
psig	pounds per square inch gauge
PTS	pressurized thermal shock (10 CFR 50.61)
PWR	pressurized water reactor
RAB	reactor auxiliary building
RCCA	rod control cluster assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RT_{NDT}	reference temperature, nil ductility transition
RT_{PTS}	reference temperature, pressurized thermal shock
RTD	resistance temperature detector

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Acronym List (continued)

RTP	rated thermal power
RTS	reactor trip system
RWST	refueling water storage tank
SAL	safety analysis limit
SBLOCA	small break loss of coolant accident
SBO	station blackout
SER	Safety Evaluation Report
SFP	spent fuel pool
SG	steam generator
SGTR	steam generator tube rupture
SRP	Standard Review Plan
SW	service water
TEDE	total effective dose equivalent
TS	Technical Specification
UHS	ultimate heat sink
V&V	verification and validation
X/Q	radiological atmospheric dispersion factor

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Introduction

This enclosure contains the HNP responses to the NRC Regulatory Issue Summary 2002-03, requested information for MUR power uprates. The LAR enclosure sections match the NRC Regulatory Issue Summary 2002-03 sections for ease of review.

I. FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

1. A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique. The description should include:

   A. Identification (by document title, number, and date) of the approved topical report on the feedwater flow measurement technique

   B. A reference to the NRC's approval of the proposed feedwater flow measurement technique

   C. A discussion of the plant-specific implementation of the guidelines in the topical report and the staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique

   D. The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique

   E. A calculation of the total power measurement uncertainty at the plant , explicitly identifying all parameters and their individual contribution to the power uncertainty

   F. Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:

      i. maintaining calibration

ii. controlling software and hardware configuration

iii. performing corrective actions

iv. reporting deficiencies to the manufacturer

v. receiving and addressing manufacturer deficiency reports

G. A proposed allowed outage time for the instrument, along with the technical basis for the time selected

H. Proposed actions to reduce power level if the allowed outage time is exceeded, including a discussion of the technical basis for the proposed reduced power level

RESPONSE TO I. FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

I.1 Detailed Description of the Shearon Harris Nuclear Power Plant, Unit No.1, Implementation of the Feedwater Ultrasonic Flow Meter

The HNP feedwater ultrasonic flowmeter is a Cameron LEFM CheckPlus ultrasonic multi-path, transit time flowmeter. This equipment also provides a highly accurate feedwater temperature that will be input to the secondary heat balance. This advanced flow measurement system design is described in detail by the manufacturer in Topical Reports ER-80P, Revision 0 (Reference I-1) and ER-157P, Revision 8 (Reference I-2).

The LEFM CheckPlus System consists of an electronic cabinet installed in the Secondary Sampling Equipment Enclosure located in the Turbine Building, and measurement spool pieces installed in each of the three main feedwater flow lines. Spool pieces in the A and B feedwater lines are installed well downstream of the existing venturis, and will have no impact on venturi performance. The spool piece in the C feedwater line is installed upstream of the venturi and will have no impact on that venturi's performance. The LEFMs were calibrated at the Alden Research Laboratory facility using the current plant piping configuration and variations of the plant configuration. The calibration test determines the meter calibration constant, or meter factor. The meter factor provides a small correction to the numerical integration to account for fluid velocity profile specifics and any dimensional measurement errors. Parametric tests are performed to determine meter factor sensitivity to up stream hydraulics.

Each measurement section consists of 16 ultrasonic, multi-path, transit time transducers divided into two planes of eight, one dual RTD, and two pressure transmitters. Each transducer may be removed at full power conditions without disturbing the pressure boundary. These flow elements conform to the installation location requirements specified in Topical Reports ER-80P and ER-157P.

The LEFM measures the transit times of ultrasonic energy pulses traveling along chordal acoustic paths through the flowing fluid. This technology provides higher accuracy and reliability than the existing venturi flow instruments. Sound travels faster when the pulse transverses the pipe with the flow and slower against the flow due to the Doppler Effect. The LEFM uses these transit times and time differences between pulses to determine the fluid velocity. The LEFM also measures the speed of sound in water and uses this measurement to determine the feedwater temperature.

The electronic cabinet controls magnitude and sequences transducer operation; makes time measurements; and calculates volumetric flow, temperature and mass flow. The system software employs the ultrasonic transit time method to measure velocities at precise locations. The system numerically integrates the measured velocities. System software has been developed and maintained under a V&V program. The feedwater mass flow rate and temperature are displayed on the electronic cabinet and transmitted to the plant process computer for use in the calorimetric measurement (secondary plant energy balance) of reactor thermal output. The system utilizes continuous calorimetric power results transmitted by direct, redundant links to the plant computer, and incorporates self-verification features. These features ensure that system performance is consistent with the design basis.

The system has two operating modes: Normal and Maintenance. Normal operation is also known as CheckPlus mode. In this mode, both planes of transducers are in service and system operations are processed by both CPUs. If the system is subjected to a failure involving a transducer, failure of one plane of operation or if a CPU related malfunction occurs, the system reverts to the Maintenance mode also known as Check mode. When a plane of operation is lost, the system alerts the control room operators through the annunciator window for Computer Alarm Reactor, and shifts from Normal operation to Maintenance mode. If the system suffers a loss of AC power or other total failure, the system also alerts the operators through the aforementioned annunciator. Operations personnel are alerted to system trouble through annunciator window Computer Alarm Reactor if the electronic cabinet internal temperature is high or when other trouble conditions occur as determined by the plant computer.

In Normal mode, the improved measurement accuracy for feedwater mass flow and temperature and a change in the way instrument uncertainty is combined for other parameters (e.g., steam pressure) results in a total uncertainty of 0.336% (rounded up to 0.34%) at RTP. This is more accurate than the nominal 2% RTP used in the accident analyses or the uncertainty currently obtainable with precision, venturi-based instrumentation and RTDs.

A new calorimetric calculation was added to the plant computer for the LEFM indications of feedwater mass flow and temperature. It runs in parallel with the venturi-based flow and RTD temperature inputs currently used in the plant calorimetric measurement calculations. The plant computer system calorimetric programs will add a correction factor to the venturi calculation, to normalize the flow and temperature to the LEFM values. Both plant computer system calorimetric programs will receive data from SG blowdown flow to calculate calorimetric power. The existing venturi-based flow and RTD temperature will continue to be used for feedwater control and other functions, and may be used for plant calorimetric measurement when the LEFM is inoperable.

I.1.A Caldon Topical Reports Applicable to the LEFM CheckPlus System

The referenced Topical Reports are:

1. ER-80P, Revision 0, *Improving Thermal Power Accuracy and Plant Safety While Increasing Operator Power Level Using LEFM Check System*, March 1997

   (Reference I-1)

2. ER-157P, Revision 8, *Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System,* August 2010 (Reference I-2)

I.1.B NRC Approval of Caldon LEFM CheckPlus System Topical Reports

The NRC approved the Topical Reports listed in I.1.A above on the following dates:

1. ER-80P, NRC SER dated March 8, 1999 (Reference I-3)
2. ER-157P, NRC SER dated August 16, 2010 (Reference I-4)

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The NRC performed an additional evaluation on the acceptability of the Cameron LEFM. The evaluation results are documented in Reference I-5, which addressed the hydraulic aspects of Cameron's LEFMs in response to industry operating experience. The NRC staff concluded that the Cameron LEFM Check and CheckPlus performance was consistent with the Topical Reports ER-80P, Revision 0 and ER-157P, Revision 5, previously approved by the NRC staff.

I.1.C HNP Implementation of Guidelines and NRC SER for the Cameron LEFM CheckPlus System

The LEFM CheckPlus System is permanently installed per the requirements specified in Topical Reports ER-80P and ER-157P. The system will be used for continuous calorimetric power determination by direct, redundant links with the plant computer. The system incorporates self-verification features to ensure that the hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis.

The plant computer system software will continuously adjust the venturi flow coefficients and the feedwater RTD temperatures to the more accurate LEFM values. The feedwater flow values for the new normalized filtered feedwater venturi flow and normalized one minute average feedwater venturi flow will be normalized to equal the LEFM feedwater flow. Normalization is performed on a loop basis. The feedwater temperature values used to determine densities for the new normalized filtered feedwater venturi flow and normalized one minute average feedwater venturi flow are based on normalized feedwater RTD temperatures biased to equal the LEFM feedwater temperatures. The calorimetric calculation is not sensitive to changes in feedwater pressure, so the venturi feedwater pressure inputs are not normalized to the LEFM feedwater pressure.

The HNP LEFM CheckPlus System was calibrated in a site-specific model test at Alden Research Labs, with traceability to National Standards. A copy of the Alden Research Labs certified calibration report is contained in the Cameron Meter Factor Report (Enclosure 6). The LEFM CheckPlus System installation and commissioning was performed according to Cameron procedures. These procedures include verification of transducer signal quality and hydraulic performance as compared to site-specific model testing at Alden Research Labs.

I.1.D Disposition of NRC SER Criteria During Installation

In approving Topical Reports ER-80P and ER-157P, the NRC established four criteria each licensee referencing these Topical Reports must address. The four criteria are listed below along with a discussion of how HNP will satisfy them.

I.1.D.i NRC Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

I.1.D.i.a Response to NRC Criterion 1

License amendment implementation will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing and training at the uprated power level using the LEFM CheckPlus System. A preventive maintenance program will be developed for the LEFM based on the vendor's maintenance and troubleshooting manual. Work on the LEFM will be performed by site instrumentation and control personnel qualified per the HNP Instrumentation & Control Training Program. The HNP Nuclear Information Technology group will assist when computer hardware or software maintenance is required.

The preventive maintenance activities include:

- General terminal and cleanliness inspection
- Power supply inspection
- Central Processing Unit inspection
- Acoustic Processor Unit checks
- Analog input/output checks
- Alarm Relay checks
- Watchdog Timer checks that ensure the software is running
- Communication checks
- Transducer checks
- Calibration checks on each feedwater pressure transmitter

The preventive maintenance program and continuous self-monitoring ensure that the LEFM remains bounded by the Topical Report ER-80P analysis and assumptions. Establishing and continued adherence to these requirements assures that the LEFM CheckPlus System is properly maintained and calibrated.

Contingency plans for plant operation with an inoperable LEFM are described in Sections I.1.G and I.1.H below.

I.1.D.ii NRC Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system, and bounds the analysis and assumptions set forth in Topical Report ER-80P.

I.1.D.ii.a Response to NRC Criterion 2

LEFMs were installed in HNP during the Fall 2010 Refueling Outage, with commissioning and calibration completed in November, 2010. Active monitoring has been ongoing since that time. The LEFM feedwater flow and temperature data have been compared to the venturi feedwater flow and feedwater RTD output. The data comparison demonstrated that the LEFM is consistent with the venturi feedwater flow and RTD feedwater temperature. There have been no maintenance related activities since LEFM installation. The HNP LEFMs are functioning as designed.

I.1.D.iii NRC Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both venturi and the LEFM for comparison.

I.1.D.iii.a Response to NRC Criterion 3

HNP uses a core thermal power uncertainty calculation approach consistent with ISA-RP67.04.02-2000 (Reference I-6) and Topical Report ER-80P (Reference I-1), as supplemented by ER-157P (Reference I-2). The methodology is described in Westinghouse WCAP-12340 (Reference I- 7), which is referenced in FSAR Section 7.7. The methodology is also used in HNP calculation HNP-I/INST-1010 (Reference I-8). Revision 0 of this calculation was submitted to the NRC in May 2001 (Reference I-9), as part of a response to a request for additional information related to SG replacement. The NRC evaluated the proposed trip setpoint values and allowable values using this calculation.

The NRC issued HNP Amendment 126 on August 31, 2007 (Reference I-10) that increased the statistical summation error term Z and the allowable value for SG trip setpoints used in the RTS and ESFAS. Calculation HNP-I/INST-1010, Revision 3 was submitted to the NRC as part of a response to a request for additional information related to SG water level trip setpoints (Reference I- 11).

The methodology used to calculate the LEFM uncertainty is based on accepted plant setpoint methodology. An alternate methodology for calculating LEFM uncertainty was not used. The fundamental approach used in the setpoint methodology is to statistically combine inputs to determine the overall uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are arithmetically combined to form statistically independent groups, which are then combined using the square root of the sum of the squares approach to determine the overall uncertainty. The same fundamental approach was used to determine the LEFM based power calorimetric uncertainty. This approach has been approved by the NRC in Topical Reports ER-80P and ER-157P and for Calvert Cliffs Units 1 & 2 (Reference I-12), North Anna Units 1 & 2 (Reference I-13) and Surry Units 1 and 2 (Reference I-14).

I.1.D.iv NRC Criterion 4

For plants where the LEFM was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), provide additional justification for use. This justification should show either that the meter installation is independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated LEFM, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

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I.1.D.iv.a Response to NRC Criterion 4

LEFMs were installed in HNP during the Fall 2010 Refueling Outage. A LEFM bounding uncertainty has been provided for use in the uncertainty calculation described below (Reference I-15). The bounding calibration factor acceptability for the spool pieces was established by tests at the Alden Research Labs (Reference I-16). These tests included a full-scale model of the HNP hydraulic geometry and a straight pipe. Test results were evaluated and documented in an Alden Research Labs test data report and Cameron engineering report. The calibration factor used for the LEFM is based on these reports. The spool piece calibration factor uncertainty is based on the Cameron engineering reports. The site-specific uncertainty analysis documents these analyses and will be maintained as part of the HNP technical basis for the power uprate.

The commissioning process verifies bounding calibration test data and provides final positive confirmation that actual field performance meets the uncertainty bounds established for the instrumentation. A Cameron installation and setup review confirmed that the HNP LEFM CheckPlus System meets the requirements specified in Cameron Report ER-697, Revision 2 (Reference I-15). Final commissioning was completed in November, 2010.

I.1.E Total Power Measurement Uncertainty at HNP

The overall thermal power uncertainty using the LEFM in Normal mode is 0.336% at RTP. The overall thermal power uncertainty using the LEFM in Maintenance mode is 0.482% at RTP. The uncertainty calculations are documented in Reference I-15, which is a Cameron proprietary document (Enclosure 5). The key parameters and their uncertainty are summarized in Table I-1. In addition to the calorimetric inputs provided by the LEFM for determination of feedwater mass flow rate and enthalpy, the HNP plant computer uses several process inputs (e.g., steam pressure, steam generator blowdown flow) to calculate the contribution of steam enthalpy and other gains and losses. These parameters are identified as Items 21 and 22 in Table I-1. For comparison, baseline values from Topical Report ER-157P, Revision 8 (Reference I-2) for the Normal mode of operation are shown in Table I-1. Differences between the HNP Normal mode uncertainties and those from ER-157P, Revision 8 are a result of plant-specific calculations and parameter uncertainties.

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Table I-1
Total Thermal Power Uncertainty Determination for HNP

Item	Parameter [1]	ER-157P, Rev. 8 Normal Mode Uncertainty	HNP Normal Mode Uncertainty (ER-697, Rev.2)	HNP Maintenance Mode Uncertainty (ER-697, Rev.2)
1	Hydraulics: Profile Factor	0.22%	0.18%	0.38%
	Geometry:			
2	Spool dimensions	0.06%	0.11%	0.11%
3	Spool piece alignment	0	0	0
4	Transducer location/replacement	0.09%	0.08%	0.12%
5	Spool piece thermal expansion, material properties	0.07%	0.08%	0.08%
6	Spool piece thermal expansion, temperature	0	0	0
	Time Measurements			
7	Time of flight measurements	0.06%	0.15%	0.15%
8	Non-fluid delay	0.043%	0.02%	0.02%
9	Volumetric flow uncertainty	0.266%	0.283%	0.447%
10	Temperature correlations	0.5°F [2]	0.5°F	0.5°F
11	Temperature/spool piece dimensions	0.04°F [2]	0.03°F	0.03°F
12	Temperature/time of flight	0.15°F [2]	0.18°F	0.18°F
13	Temperature/pressure	0.21°F [2]	-0.21°F	-0.21°F
	Feedwater Density [3][4]			
14	Feedwater density/ASME correlation	0.04%	0.04%	0.04%
15	Feedwater density/temperature	0.05%	-0.05%	-0.05%
16	Feedwater density/pressure	0.01%	0.03%	0.03%
	Feedwater Enthalpy [3][4]			
17	Feedwater enthalpy/temperature	0.08%	-0.07%	-0.07%
18	Feedwater enthalpy/pressure	Nil	0.03%	0.03%
19	Power uncertainty, thermal expansion	0.04%	0.04%	0.04%
	Steam Enthalpy:			
20	Steam enthalpy/moisture	0	0	0
21	Steam enthalpy/pressure	0.07%	0.0534%	0.0534%
22	Gains/Losses	0.07%	0.0456%	0.0456%
23	Total Thermal Power Uncertainty	0.396%	0.336%	0.482%

Notes:
1. Items 1 through 19 are directly associated with the LEFM. Items 20 through 22 are based on other plant process inputs.
2. The total uncertainty in temperature, as determined by the LEFM, is the root sum square of the individual contributions delineated on lines 10, 11, 12, and 13. The result is 0.565°F for LEFM CheckPlus.
3. Total power uncertainty is = RSS (Items 1,2,3,4,5 + 19,6,7,8,14,15 + 17,16 + 18,20,21,22.
4. The bounding uncertainties in pressure and temperature are +15 psi and +0.57°F, respectively.

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The uncertainty for transducer installation, as identified in Caldon Customer Information Bulletin CIB-125 (Reference I-17), has been included in the HNP LEFM uncertainty (Reference I-15). These system uncertainties incorporate an additional transducer variability uncertainty in both the profile factor uncertainty and in the installation uncertainty.

I.1.F Calibration and Maintenance Procedures of Instruments Affecting the Power Calorimetric

Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric.

I.1.F.i Maintaining Calibration

LEFM hardware and instrumentation calibration and maintenance will be performed using procedures based on the appropriate Cameron LEFM CheckPlus System technical manuals, which ensures that the LEFM remains bounded by the Topical Report ER-80P analysis and assumptions. The other calorimetric process instrumentation and computer points are maintained and periodically calibrated using approved procedures. Preventive maintenance tasks are periodically performed on the plant computer system and support systems to ensure continued reliability. Work is planned and executed in accordance with established HNP work control processes and procedures. Routine LEFM preventive maintenance activities will include, but not be limited to, those activities specified in Section I.1.D.i.a.

I.1.F.ii Controlling Software and Hardware Configuration

The LEFM CheckPlus System is designed and manufactured per Cameron's 10 CFR 50, Appendix B, Quality Assurance Program and V&V Program. Cameron's V&V Program fulfills the requirements of ANSI/IEEE-ANS Standard 7-4.3.2, 1993 (Reference I-18) and ASME NQA-1a-1999, Subpart 2.7 (Reference I-19). After installation, the LEFM software configuration will be maintained using existing procedures and processes, which include V&V of software configuration changes. LEFM hardware and the calorimetric process instrumentation will be maintained per the HNP configuration control processes.

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I.1.F.iii Performing Corrective Actions

Plant instrumentation that affects the power calorimetric, including the LEFM inputs, will be monitored by HNP personnel. Problems detected are documented per the HNP corrective action program, with necessary follow-up actions planned and implemented.

I.1.F.iv Reporting Deficiencies to the Manufacturer

Conditions found to be adverse to quality (as defined in 10 CFR 50, Appendix B) will be documented per the HNP corrective action program and reported to the vendor, as needed, to support corrective action.

I.1.F.v Receiving and Addressing Manufacturer Deficiency Reports

HNP has existing processes for addressing manufacturer's deficiency reports. Such deficiencies will be documented in the HNP corrective action program and actions will be controlled by the HNP work control process.

I.1.G Completion Time and Technical Basis

A completion time of 72 hours is proposed for operation at any power level above the current licensed power of 2900 MWt with the LEFM not fully functional. The basis for the proposed 72 hour completion time follows:

- Operations procedures will direct the use of the back-up calorimetric in the event of LEFM failure (Fail mode). This algorithm receives input from alternate plant instruments (feedwater venturis and RTDs) for feedwater flow rate calculation. The feedwater flow from the three venturis will be normalized to the LEFM feedwater flow rate, so that the alternate calorimetric matches the primary LEFM based calorimetric. Also, the feedwater RTD temperature measurements will be normalized to the more accurate data from the LEFM.
- HNP has performed a drift study of the feedwater flow transmitters. As found/as left calibration data from May 6, 2006 through November 8, 2010 was obtained for all six feedwater flow transmitters used by the ERFIS calorimetric calculation. These transmitters are calibrated on a refueling interval basis, so the study included four complete calibrations for each transmitter and 27 drift data points per transmitter. The

results indicate the worst case transmitter drift over the 18 month calibration interval is 0.45%. Conservatively assuming that all six feedwater flow transmitters (two per loop) drifted by this magnitude in the same direction, the impact on thermal power measurement over the proposed 72-hour completion time has been calculated as less than 0.1 MWt. This assumes a linear drift behavior over the 18 month interval. Based on the calculated average drift value for all feedwater transmitters over the 18 month interval, the impact on thermal power measurement over the 72-hour period would be negligible.

- One feedwater flow venturi is visually inspected each refueling outage. No venturi fouling has been observed to date. Based on these inspection results, it is very unlikely that venturi fouling or defouling would occur during the proposed 72-hour completion time.
- LEFM repairs are expected to be completed within an 8-hour shift. A completion time of 72 hours provides plant personnel sufficient time to diagnose, plan and package work orders, complete repairs, and verify normal system operation within original uncertainty bounds.

The 72-hour completion time begins when the annunciator alarm is received in the main control room. A control room alarm response procedure will be developed providing guidance to the operators for initial alarm diagnosis. Methods to determine LEFM CheckPlus System status and the cause of alarms are described in Cameron documentation. Cameron documentation will be used to develop specific procedures for operators and maintenance response actions.

I.1.H Actions for Exceeding Completion Time and Technical Basis

The Cameron LEFM CheckPlus system has two operating modes (Normal and Maintenance) and a Fail mode.

- Normal: The LEFM CheckPlus System Normal is displayed when all the feedwater flow, temperature, and header pressure signals for feedwater loops A and B are normal and operating within design limits. Calculated power level uncertainty associated with the LEFM flow measuring system in this condition is 0.34%.
- Maintenance: A LEFM CheckPlus System Alert alarm indicates a loss of system redundancy and the system shifts from the Normal mode to the Maintenance mode of operation. The calculated power level uncertainty associated with the LEFM flow measuring system in this condition is 0.48%. An Alert alarm is caused by:

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1. Loss of a single process input
    a. Loss of a single flow plane (loss of one or more flow transducers in a flow plane) on any feedwater line
    b. Loss of a single flow plane (loss of one or more flow transducers in a flow plane) in multiple feedwater lines
    c. Loss of a single redundant spool piece RTD on any line
    d. Loss of a single redundant feedwater header pressure input
2. Loss of a single Electronics Unit redundant component. The Electronics Unit includes two redundant systems, each one includes a separate power supply with 5 volt, +12 volt, -12 volt, 24 volt and 180 volt outputs; four acoustical signal processing units that transmit and receive ultrasonic flow signals; and a CPU that performs flow and temperature calculations, system self checks, and system verifications. The loss of any one of these components would produce an Alert alarm.
3. Process input or output is calculated outside a pre-determined allowable range.
4. Internal self-check indicates system parameters that exceed pre-established limits and affect a single plane in one or more loops; for example, problems could be identified with the Global Synchronization Signal board, signal Rejects, signal Transit Time, path high gain, or speed of sound.

- Fail: A LEFM CheckPlus System Fail alarm indicates a loss of function and the power level uncertainty reverts to the 2.0% error associated with the venturi flow meters. A Fail alarm is caused by:

1. Loss of both redundant process inputs
    a. Loss of both planes (A & B) on a single feedwater line or multiple feedwater lines
    b. Loss of both redundant spool piece RTDs on a single loop
    c. Loss of both feedwater header pressure inputs
2. Failure of both redundant components in the Electronics Unit, such as both 180 volt power supplies.
3. A process input or output is calculated outside a pre-determined allowable range by both CPU units.

4. Loss of the data link between the CheckPlus system and the plant computer.

5. Internal self-check indicates system parameters that exceed pre-determined limits and affect multiple planes in one or both loops; for example, problems could be identified with the Global Synchronization Signal board, signal Rejects, signal Transit Time, path high gain, or speed of sound.

I.1.H.i Cameron LEFM - Fail Mode

In the event the LEFM is inoperable (Fail mode), the feedwater flow rate and feedwater temperature inputs to the calorimetric will be determined by alternate instrumentation. The existing feedwater venturi flow nozzles and RTDs will be used for the calorimetric until the LEFM is returned to operable status. To ensure that the venturi-based calorimetric is consistent with the LEFM based calorimetric, the venturi-based flow rate and feedwater temperature will be normalized to the LEFM. A plant computer loss is treated as a loss of both the LEFM and the ability to obtain corrected calorimetric power using the alternate plant instrumentation. Operation with a plant computer loss at the uprated power level may continue until the next required nuclear instrumentation heat balance, which could be up to 24 hours. A plant computer failure will require reducing core thermal power to 2900 MWt as needed to support a manual calorimetric power calculation. These requirements ensure that an operable low uncertainty (< 2.0%) input is used whenever core power is greater than 2900 MWt. With the LEFM in Fail mode, if the plant experiences a power decrease below 2900 MWt (98.4% of RTP) during the 72 hour completion time (allowed outage time), the maximum permitted power level will be the current licensed core power level of 2900 MWt until the LEFM is restored to either Normal or Maintenance mode operation. It is conservative to limit the power level to 98.4% RTP until the LEFM is returned to functional status. The operators will be provided with procedural guidance for those occasions when the LEFM is inoperable.

I.1.H.ii Cameron LEFM - Maintenance Mode

As stated above, a single path or plane malfunction (Maintenance mode) results in an uncertainty change from 0.34% to 0.48% (0.14% difference). In the event of a failure of one path or plane that cannot be restored to full functionality (Normal mode) within 72 hours, power will be reduced to approximately 99.86% RTP (2943 MWt, rounded down). The plant can operate at this power level indefinitely with a single plane of LEFM system. The operators will be provided with procedural guidance for those occasions when the LEFM is in the Maintenance mode.

Topical Report ER-157P, Revision 8 (Reference I-2) states that the redundancy inherent in the two measurement planes of an LEFM CheckPlus System also makes this system more resistant to component failures when compared to the LEFM Check System. For any single component failure, continued operation at a power greater than that prior to the uprate can be justified with the LEFM CheckPlus System since the system with the failure is no less than an LEFM Check. The NRC SER (Reference I-4) approving ER-157P, Revision 8 required licensees referencing ER-157P, Revision 8 to ensure compliance with two limitations and conditions:

- Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.
- The only mechanical difference that potentially affects the Topical Report ER-157P, Revision 8 statement above is that the LEFM CheckPlus System has 16 transducer housing interfaces with the flowing water, whereas the LEFM Check System has 8. Consequently, a LEFM CheckPlus System operating with a single failure that is assumed to disable one plane of transducers is not identical to an LEFM Check System. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate as stated. An acceptable quantification method is to establish the effect in an acceptable test configuration such as can be accomplished at the Alden Laboratory.

HNP has provided plant-specific justification for the proposed completion time in Section I.1.G above. The impact of a single failure (Maintenance mode) on the HNP CheckPlus System has been quantified as an uncertainty of 0.48%. Cameron ER-697, Revision 2 (Enclosure 5), identifies the uncertainties associated with LEFM operation in the CheckPlus and Check (Maintenance) modes, including meter factor uncertainties specific to HNP. These uncertainties are provided in Table I-1 above and were established by the calibration tests performed at Alden Research Laboratory. The associated change in uncertainty from 0.34% to 0.48% (0.14% difference) justifies the proposed power reduction to 99.86% for this condition. HNP has satisfied the two limitations and conditions specified in the NRC SER for licensees referencing Topical Report ER-157P, Revision 8.

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I.1.H.iii Cameron LEFM - Operability Requirements

The LEFM operability requirements will be contained in the HNP Relocated TS and Design Basis Requirements procedure PLP-114 (Reference I-20). The PLP-114 procedure is incorporated by reference in FSAR Section 13.5.1. The proposed LEFM operability requirements are provided in Attachment 4 for information only.

I REFERENCES

I-1 Caldon Engineering Report ER-80P, Revision 0, *Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System,* Caldon Inc., March 1997.

I-2 Cameron Engineering Report ER-157P, Revision 8, *Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System,* Caldon Inc., May 2008.

I-3 Letter from Project Directorate IV-1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, to C.L. Terry (TU Electric), Comanche Peak Steam Electric Station, Units 1 and 2 - *Review of Caldon Engineering Topical Report 80P, Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System* (Accession Number 9903190065 legacy library), March 8, 1999.

I-4 Letter from Thomas B. Blount (USNRC) to Ernest Hauser (Cameron), *Final Safety Evaluation For Cameron Measurement Systems Engineering Report ER-157P, Revision 8, Caldon Ultrasonics Engineering Report ER-157P, Supplement to Topical Report ER-80P: Basis For A Power Uprate With The LEFM Check Or CheckPlus System*, ML102160663, August 16, 2010.

I-5 Letter from Brian E. Thomas (USNRC) to E.M. Hauser (Caldon, Inc.), *Evaluation of the Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus Ultrasonic Flow Meters (UFM)* ML061700222, July 5, 2006.

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I-6 ISA-RP67.04.02-2000, *Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation.*

I-7 WCAP-12340, Revision 1, *Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for Carolina Power & Light Harris Nuclear Plant (For Uprate to 2912.4 MWT-NSSS Power and Replacement Steam Generator),* January 2000.

I-8 HNP Calculation HNP-I/INST-1010, Revision 4, *Evaluation of Tech Spec Related Setpoints, Allowable Values, and Uncertainties Associated with RTS/ESFAS Functions for Steam Generator Replacement (with Current 2787 MWT-NSSS Power or Uprate to 2912.4 MWT-NSSS Power),* February 21,2011.

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I-10 Letter from Marlayna Vaaler (USNRC) to Robert J. Duncan II (Carolina Power & Light Company), *Shearon Harris Nuclear Power Plant, Unit 1 - Issuance of Amendment to Revise Steam Generator Water Level Trip Setpoint Values (TAC No. MD2723),* August 31, 2007.

I-11 Letter from D. H. Corlett (Progress Energy Carolinas) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant, Unit No. 1, Docket No. 50-400/License No. NPF-63, Response to Request for Additional Information (RAI) Regarding the License Amendment Request to Revise Values Associated with the Steam Generator Water Level (SGWL) Trip Setpoints,*HNP-07-033, March 9, 2007.

I-12 Letter from Douglas V. Pickett (USNRC) to James A. Spina (Calvert Cliffs Nuclear Power Plant Inc.), *Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment RE: Measurement Uncertainty Recapture Power Uprate,* ML091820366, July 22, 2009.

I-13 Letter from V. Sreenivas (USNRC) to David A. Heacock (Virginia Electric and Power Company), *North Anna Power Station, Unit Nos. 1 and 2 - Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML092250616, October 22, 2009.

I-14 Letter from Karen Cotton (USNRC) to David A. Heacock (Virginia Electric and Power Company), *Surry Power Station, Unit Nos. 1 and 2 - Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML101750002, September 24, 2010.

I-15 Cameron Ultrasonics Engineering Report ER-697, Revision 2, *Bounding Uncertainty Analysis for Thermal Power Determination at Harris Unit 1 Using the LEFM CheckPlus System*, January 2011.

I-16 Cameron Ultrasonics Engineering Report ER-720, Revision 2, *Meter Factor Calculation and Accuracy Assessment for Harris Nuclear Plant*, January 2011.

I-17 Caldon *Customer Information Bulletin CIB-125*, Revision 0, April 23, 2007.

I-18 ANSI/IEEE-ANS Standard 7-4.3.2, 1993, *IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations.*

I-19 ASME NQA-1a-1999, *Quality Assurance Requirements for Nuclear Facility Applications.*

I-20 PLP-114, *Relocated Technical Specifications and Design Basis Requirements*, Revision 21.

II. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

1. A matrix that includes information for each analysis in this category and addresses the transients and accidents included in the plant's updated final safety analysis report (UFSAR) (typically Chapter 14 or 15) and other analyses that licensees are required

to perform to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses to determine environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding):

A. Identify the transient or accident that is the subject of the analysis

B. Confirm and explicitly state that

   i. the requested uprate in power level continues to be bounded by the existing analyses of record for the plant

   ii. the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC

C. Confirm that bounding event determinations continue to be valid

D. Provide a reference to the NRC's previous approvals discussed in Item B. above

RESPONSE TO II. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

II.1 Introduction

A review of FSAR (Reference II-1) Chapters 6 and 15 and other related subsections was performed to support the MUR power uprate with respect to the accident analyses. Evaluations were also performed on other analyses (e.g., internal flooding, station blackout, natural circulation cooldown). The FSAR review was conducted to confirm that the existing analyses of record, as currently presented in the FSAR, were performed conservatively and remain valid and bounding for the proposed power uprate.

The analyses generally model the core and/or NSSS thermal power in one of three ways. First, some analyses apply a 2.0% increase to the initial power level to account for the power measurement uncertainty. These analyses have not been re-performed for the MUR uprate conditions, because the sum of the proposed core power level and the decreased power measurement uncertainty falls within the previously analyzed conditions. The existing 2.0% uncertainty is reallocated so a portion is applied to uprate power and the remainder is retained to accommodate the power measurement uncertainty. Second, some analyses employ a nominal power level. These analyses have either been evaluated or re-performed for the proposed power level. Third, some of the analyses are performed at 0% power conditions or do not actually model core power level. These analyses have not been re-performed because they are unaffected by the core power level.

For the HNP MUR power uprate, a core RTP of 2948 MWt was selected based on the calorimetric uncertainty of 0.34% (rounded up from 0.336) with the LEFM and a review of the accident analysis assumptions for core power. The deterministic accident analyses use 2958 MWt (102% of 2900 MWt) as the total core power, which leaves 10 MWt of margin to accommodate the power uncertainty. The 10 MWt represents 0.339% of 2948 MWt. Since the power calorimetric uncertainty of 0.336% at RTP with the LEFM is less than the accident analysis allowance of 0.339% with a 2948 MWt licensed power level, the deterministic accident analyses are bounding for the MUR power uprate.

II.1.A Fuel Evaluation - Zircaloy 4 Cladding

II.1.A.i Fuel Assembly Mechanical Performance

The fuel design was evaluated to determine the power uprate impact on the fuel assembly design criteria. The evaluation concluded that the fuel design remains acceptable and continues to satisfy the required design criteria with the uprate operating temperature, pressure, and flow rates. The evaluation methodology compared the significant operating parameter values used in the analysis of record, with the values proposed for the power uprate. Significant parameters evaluated included: inlet temperature, system pressure, core average linear heat generation rates, maximum fuel rod and assembly axial average fluence, minimum coolant flow rate, fuel residence time, and peak fuel rod burnup. These parameters affect important design criteria such as: fuel rod cladding strain, fatigue, oxidation, collapse, fuel rod gas pressure, fuel assembly hold-down margin, and shoulder gap. The significant parameter evaluation showed that the proposed power uprate operating and transient values are still justified. Therefore, the fuel mechanical performance design criteria are satisfied under MUR power uprate conditions.

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II.1.A.ii Nuclear Design

The power uprate impact on the nuclear core design was evaluated using HNP Cycle 17 (which began in the 4th quarter 2010), as a representative fuel cycle. The range of parameters used in the current FSAR Chapter 15 safety analyses of record are adequate to accommodate the range of parameters expected for future cores operating at the uprated power level. Simulated core reload analyses, for future uprated core cycles, demonstrated that MUR implementation does not result in significant changes to the current nuclear design basis for the analysis documented in the FSAR. The evaluation also concluded that no unique changes to the physical design of the reactor core, fuel assemblies, or core components are required to implement to MUR power uprate.

The impact on core peaking factors, control rod worth, reactivity coefficients, shutdown margin, and kinetics parameters were within normal cycle-to-cycle variation of these values or controlled by the nuclear core design. Future impacts will be addressed on a cycle-specific basis consistent with approved reload methodologies.

The nuclear design methods and core simulator models used to evaluate the power uprate are consistent with those listed in the TS. There are no changes to currently approved nuclear design methodologies, nuclear design philosophy or core physics models/codes. Core design and power distribution control will continue to be performed using the approved methodologies specified in TS 6.9.1.6.

II.1.A.iii Fuel Rod Design

The AREVA 17x17 HTP fuel rod design was evaluated to determine the power uprate impact on the generically approved fuel rod design criteria in Reference II-2. The evaluation concluded that the fuel rod design remains acceptable and continues to satisfy the required design criteria up to AREVA's peak rod average burnup limit of 62,000 MWD/MTU, when the fuel is operated within the peaking limits specified in the TS.

The evaluation methodology compared the proposed plant operating parameters at power uprate conditions with the corresponding parameters used in the analysis of record for Cycle 17. The impact of the plant operating parameter changes on the fuel rod design with Zircaloy-4 cladding was evaluated using the approved mechanical design methods in References II-3 and II-4. The evaluation results showed that all fuel rod design criteria were satisfied up to AREVA's fuel rod burnup licensing limit.

II.1.A.iv Core Thermal-Hydraulic Design

The power uprate impact on the core thermal-hydraulic design was considered in evaluations that examined the Chapter 15 transients with respect to DNB, setpoint analyses, and core safety limit lines. The evaluation of events that challenge the DNBR acceptance criteria concluded that the existing analyses of record for deterministic cases, based on currently approved methods and codes (References II-5 through II-10), satisfactorily protect the uprated condition, because those analyses were performed at 2958 MWt or 102% of 2900 MWt. Several of the Cycle 17 DNB events could not be evaluated without recalculation, because they used a statistical method. The statistical method has power uncertainty as an input. These events were evaluated and the results indicated that the DNB acceptance criteria were met for all cases at power uprate conditions.

The statistical evaluation of current plant setpoints (OTΔT and OPΔT), using currently approved methods and codes (Reference II-10), takes into account the higher nominal power, reduced uncertainty, and design axial shapes at uprate conditions. This evaluation concluded that the existing setpoints adequately protect the plant from DNB, hot leg saturation, and fuel centerline melt.

A statistical evaluation of the core safety limit lines, using the currently approved methods and codes (Reference II-10), took into account the higher nominal power, reduced uncertainty, and design axial shapes at uprate conditions. The evaluation concluded that the existing core safety limit lines (TS Figure 2.1-1) should be modified for consistency with the plant setpoints evaluation.

Cycle-specific evaluations of plant DNB transients, setpoints (OTΔT and OPΔT), and core safety limit lines are completed during the development of each core reload design. Those evaluations will include the power uprate with the final set of cycle-specific inputs.

II.1.B Fuel Evaluation - M5™ Cladding

PEC has submitted both a license amendment request (Reference II-42) and an associated exemption request (Reference II-46) from the requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K, to allow the use of M5™ advanced fuel rod cladding at HNP. M5™ is a zirconium alloy composed of zirconium and niobium (1% nominal). M5^{TM} has demonstrated superior corrosion resistance and reduced creep as compared to both standard and low-tin Zircaloy-4 cladding. M5^{TM} implementation involves changes to the fuel rod material; fuel rod diameter and thickness are unchanged. The impact of this material change on the proposed power uprate was evaluated considering fuel assembly mechanical performance, nuclear design, fuel rod design, core thermal-hydraulic design, FSAR Chapter 15 analyses, and protective system settings. The MUR power uprate does not significantly impact M5^{TM} fuel cladding implementation.

II.1.B.i Fuel Assembly Mechanical Performance

M5^{TM} fuel cladding has an insignificant impact on fuel assembly frequency response. Reference II-2 remains the applicable methodology for fuel design, as supplemented by BAW-10240(P)(A) (Reference II-43). Reference II-42 is adding methodology BAW-10240(P)(A) to the TS.

II.1.B.ii Nuclear Design

The fuel cladding change results in an alloy content shift from approximately 1.4% tin (Sn) in Zircaloy-4 to 1.0% niobium (Nb) in M5^{TM}. This change causes a slight increase in parasitic neutron absorption within the cladding material. The increase in parasitic neutron absorption only affects fuel cycle economics, and can easily be offset with minor increases in U^{235} fuel enrichment. The overall change in core reactivity between the two cladding materials is insignificant (a change in cycle energy of approximately one EFPD for an 18-month cycle). M5^{TM} cladding material has an insignificant impact on control rod worth, local pin power distributions, and axial peaking factor distributions. The conclusions in Section II.1.A.iii above remain valid for both Zircaloy-4 and M5^{TM} cladding.

II.1.B.iii Fuel Rod Design

The AREVA 17x17 HTP fuel rod design with M5^{TM} cladding was evaluated, to determine the power uprate impact on those generically approved fuel rod design criteria specified in References II-2 and II-43 that are sensitive to power level. These criteria include: cladding oxidation, fuel rod gas pressure, cladding strain, cladding collapse, cladding fatigue, and fuel rod growth. The evaluation was performed using the approved fuel rod mechanical design methods described in References II-3 and II-4. The results demonstrated that these fuel rod design criteria were satisfied up to AREVA's peak rod average burnup limit of 62,000 MWD/MTU, when the fuel is operated within the peaking limits specified in the TS.

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II.1.B.iv Core Thermal-Hydraulic Design

The M5^{TM} cladding has a small effect on the core thermal-hydraulic analyses that is within the range of impact for normal cycle to cycle variations. Any impact will be assessed using the core reload methodology. The core thermal-hydraulic analysis results supporting the power uprate are unaffected by M5^{TM} cladding.

II.1.B.v FSAR Chapter 15 Events

II.1.B.v.1 LOCA Events

The change to M5^{TM} cladding does not affect power uprate acceptability, because the applicable LOCA methodologies require a nominal reactor power plus uncertainty. As a result, the LOCA safety analysis power remains at 2958 MWt or 102% of 2900 MWt.

The SBLOCA methodology is currently XN-NF-82-49 (References II-27 and II-28). A pending license amendment request (Reference II-42) regarding M5^{TM} cladding will adopt the EMF-2328(P)(A) (Reference II-44) methodology and supersede XN-NF-82-49 for SBLOCA. EMF-2328(P)(A) is currently approved for use at HNP and is specified in TS 6.9.1.6.2.m, but has not been used as the analysis of record for SBLOCA. After implementation of the TS change, XN-NF-82-49 will no longer be applicable to HNP and will be removed as a reference from TS 6.9.1.6.2.k.

The LBLOCA methodology is currently EMF-2087(P)(A)(Reference II-26). EMF-2087(P)(A) is mechanistic and limited to material properties for Zircaloy and stainless steel cladding. An LAR (Reference II-38) was submitted to adopt the AREVA Realistic LBLOCA methodology, EMF-2103(P)(A) (Reference II-45) and add this methodology to the TS. However, the LAR was subsequently withdrawn to supplement the analysis to address generic NRC concerns with using EMF-2103(P)(A). A revised submittal will contain plant specific methodology that takes exception to EMF-2103(P)(A) in those areas where the NRC staff identified specific concerns during their review of Reference II-38. The LBLOCA methodology will use a safety analysis core power of 2958 MWt and thus be neutral relative to the MUR power uprate LAR review.

The LBLOCA and SBLOCA analyses for the fuel design with M5^{TM} cladding will be completed prior to the introduction of M5^{TM} clad fuel into the HNP core.

II.1.B.v.2 Non-LOCA Events

The conclusions in Section II.2 regarding non-LOCA events are valid for both the current fuel design with Zircaloy-4 cladding and the fuel design with M5™ cladding.

II.1.B.vi Protective System Settings

The use of M5™ fuel cladding does not affect existing protective system settings.

II.1.C Accident/Transient/Other Analyses Matrix

Table II-1 below provides a brief overview of the accident/transient analyses and other analyses contained in the FSAR (Reference II-1), including the assumed core power level in each analysis and whether these analyses remain bounding for the MUR power uprate. This table also contains a reference to the NRC's previous approval of each analysis, if applicable, or a statement that an NRC approved method was used in the analyses of record implemented under the provisions of 10 CFR 50.59. Section II.2, Discussion of Events, elaborates on each FSAR event.

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Table II-1
FSAR Accidents, Transients and Other Analyses

Accident/Transient	FSAR Section	Assumed Reactor Power Level (% of 2900 MWt)	Bounding (Yes/No)	NRC Approval
Increase in Heat Removal by the Secondary System				
Feedwater System Malfunctions that Results in a Decrease in Feedwater Temperature	15.1.1	102	Yes	This event is bounded by the Excessive Increase in Secondary Steam Flow Event (15.1.3).
Feedwater System Malfunctions that Results in a Increase in Feedwater	15.1.2	102	Yes	Analysis was performed using NRC approved methodologies.
Excessive Increase in Secondary Steam Flow	15.1.3	102	Yes	Analysis was performed using NRC approved methodologies.
Inadvertent Opening of a Steam Generator Relief or Safety Valve	15.1.4	102	Yes	This event is bounded by the Excessive Increase in Secondary Steam Flow Event (15.1.3) Mode 1 and Steam System Piping Failure Event (15.1.5) Modes 2 & 3.
Steam System Piping Failure	15.1.5	0 and 100	Yes	Analysis was performed using NRC approved methodologies.
Decrease in Heat Removal by the Secondary System				
Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	15.2.1	N/A	N/A	This is a BWR event that is not applicable to HNP.

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Table II-1
FSAR Accidents, Transients and Other Analyses (continued)

Accident/Transient	FSAR Section	Assumed Reactor Power Level (% of 2900 MWt)	Bounding (Yes/No)	NRC Approval
Loss of External Electrical Load	15.2.2	102	Yes	This event is bounded by the Turbine Trip Event (15.2.3).
Turbine Trip	15.2.3	102	Yes	Analysis was performed using NRC approved methodologies.
Inadvertent Closure of Main Steam Isolation Valves	15.2.4	102	Yes	This event is bounded by the Turbine Trip Event (15.2.3).
Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	15.2.5	102	Yes	This event is bounded by the Turbine Trip Event (15.2.3).
Loss of Non-Emergency AC Power to the Station Auxiliaries	15.2.6	102	Yes	Analysis was performed using NRC approved methodologies.
Loss of Normal Feedwater Flow	15.2.7	102	Yes	Analysis was performed using NRC approved methodologies.
Feedwater System Pipe Break	15.2.8	102	Yes	Analysis was performed using NRC approved methodologies.
Decrease in Reactor Coolant System Flow Rate				
Partial Loss of Forced Reactor Coolant Flow	15.3.1	102	Yes	This event is bounded by the Complete Loss of Forced Reactor Coolant Flow Event (15.3.2).
Complete Loss of Forced Reactor Coolant Flow	15.3.2	102	Yes	Analysis was performed using NRC approved methodologies.

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Table II-1
FSAR Accidents, Transients and Other Analyses (continued)

Accident/Transient	FSAR Section	Assumed Reactor Power Level (% of 2900 MWt)	Bounding (Yes/No)	NRC Approval
RCP Shaft Seizure (locked rotor)	15.3.3	102	Yes	Analysis was performed using NRC approved methodologies.
RCP Pump Shaft Break	15.3.4	102	Yes	This event is bounded by the RCP Shaft Seizure Event (15.3.3).
Reactivity and Power Distribution Anomalies				
Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition	15.4.1	0	Yes	Analysis was performed using NRC approved methodologies.
Uncontrolled RCCA Bank Withdrawal at Power	15.4.2	102	Yes	Analysis was performed using NRC approved methodologies.
Dropped Full Length RCCA or RCCA Bank	15.4.3.1	102	Yes	Analysis was performed using NRC approved methodologies.
Withdrawal of a Single Full Length RCCA	15.4.3.2	102	Yes	Analysis was performed using NRC approved methodologies.
Statically Misaligned RCCA or Bank	15.4.3.3	102	Yes	Analysis was performed using NRC approved methodologies.
Startup of an Inactive RCP at an Incorrect Temperature	15.4.4	N/A	N/A	Technical Specification 3.4.1.1 (Reactor Coolant System) prohibits power operation with less than three reactor coolant loops in service (Modes 1 & 2).

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Table II-1
FSAR Accidents, Transients and Other Analyses (continued)

Accident/Transient	FSAR Section	Assumed Reactor Power Level (% of 2900 MWt)	Bounding (Yes/No)	NRC Approval
Malfunction or Failure of Flow Controller in a BWR Loop that results in an Increased Reactor Coolant Flow Rate	15.4.5	N/A	N/A	This is a BWR event that is not applicable to HNP.
CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	15.4.6	102	Yes	This event is bounded by the Uncontrolled RCCA Bank Withdrawal at Power Event (15.4.2).
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	15.4.7	102	Yes	Analysis was performed using NRC approved methodologies
Spectrum of RCCA Ejection Accidents	15.4.8	102	Yes	Analysis was performed using NRC approved methodologies
Spectrum of Rod Drop Accidents in a BWR	15.4.9	N/A	N/A	This is a BWR event that is not applicable to HNP.
Increase in Reactor Coolant Inventory				
Inadvertent Operation of the ECCS During Power Operation	15.5.1	102	Yes	Analysis was performed using NRC approved methodologies

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Table II-1
FSAR Accidents, Transients and Other Analyses (continued)

Accident/Transient	FSAR Section	Assumed Reactor Power Level (% of 2900 MWt)	Bounding (Yes/No)	NRC Approval
CVCS System Malfunction that Increases Reactor Coolant Inventory	15.5.2	N/A	N/A	This event is bounded by CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (15.4.6) and Inadvertent Operation of the ECCS During Power Operation (15.5.1).
Decrease in Reactor Coolant Inventory				
Inadvertent Opening of a Pressurizer Safety or PORV	15.6.1	102	Yes	Analysis was performed using NRC approved methodologies
Break in Instrument Line or Other Line From Reactor Coolant Pressure Boundary that Penetrate Containment	15.6.2	102	Yes	NRC approved in Reference II-13
Steam Generator Tube Rupture	15.6.3	102	Yes	Analysis was performed using NRC approved methodologies
Large Break Loss of Coolant Accident	15.6.5.2	102	Yes	Analysis was performed using NRC approved methodologies
Small Break Loss of Coolant Accident	15.6.5.3	102	Yes	Analysis was performed using NRC approved methodologies
Radioactive Release from a Subsystem or Component				
Radioactive Waste Gas System Leak or Failure	15.7.1	102(1)	Yes	NRC approved in Reference II-13

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Table II-1
FSAR Accidents, Transients and Other Analyses (continued)

Accident/Transient	**FSAR Section**	**Assumed Reactor Power Level (% of 2900 MWt)**	**Bounding (Yes/No)**	**NRC Approval**
Liquid Waste System Leak or Failure	15.7.2	N/A	N/A	This event is bounded by the Postulated Radioactive Releases Due to Liquid Tank Failure (15.7.3)
Postulated Radioactive Releases Due to Liquid Tank Failure	15.7.3	102[1]	Yes	NRC approved in Reference II-13
Design Basis Fuel Handling Accidents	15.7.4	102	Yes	NRC approved in Reference II-13
Anticipated Transients Without Scram	15.8	102	Yes	NRC approved in Reference II-14

Other Analyses	**FSAR Section**	**Assumed Reactor Power Level (% of 2900 MWt)**	**Bounding (Yes/No)**	**NRC Approval**
Natural Circulation Cooldown	15.2.6	102	Yes	NRC approved in Reference II-15
Long Term LOCA Mass & Energy Release	6.2.1.3	102	Yes	NRC approved in Reference II-16.
Short Term LOCA Mass & Energy Release	6.2.1.2a	102 [2]	Yes	NRC approved in Reference II-17
Main Steam Line Break Mass & Energy	6.2.1.4	102	Yes	NRC approved in Reference II-17
Station Blackout	8.3.1.2.21	102	Yes	NRC approved in Reference II-19 and II-20.
Analysis to Determine EQ Parameters	3.11	102	Yes	NRC approved in Reference II-21
Safe Shutdown Fire Analysis	9.5.1.3 [3]	100 [4]	Yes	NRC approved in Reference II-22
Spent Fuel Pool Cooling	9.1.3	102	Yes	NRC approved in Reference II-17

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FSAR Accidents, Transients and Other Analyses (continued)

Internal Flooding	3.6A.3.2 3.6A.6	102	Yes	NRC approved in Reference II-24

Notes:
1. Based on 1% failed fuel fission product inventory in the RCS.
2. The short-term LOCA mass and energy releases are affected by changes in RCS temperatures, which are a function of core power. Evaluations compared the postulated line break mass and energy releases at power uprate conditions to the current conditions.
3. FSAR Section 9.5.1.3 describes the fire hazard safe shutdown analyses. The analyses are maintained in engineering calculations.
4. Conservative assumption because NFPA 805 allows for the plant to be in any condition from 100% to Mode 4.

II.2 Discussion of Events

FSAR Chapter 15 accidents/transients and other FSAR analyses were reviewed to support the MUR power uprate. A summary of each evaluation is provided below.

II.2.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature - FSAR 15.1.1

This event is bounded by the Excessive Increase in Secondary Steam Flow event (FSAR 15.1.3) per the methodology in Reference II-6.

II.2.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow - FSAR 15.1.2

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.3 Excessive Increase in Secondary Steam Flow - FSAR 15.1.3

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve - FSAR 15.1.4

This event is bounded by the Excessive Increase in Secondary Steam Flow event (FSAR 15.1.3) at power. After reactor trip, this event is bounded by the Steam System Piping Failure event (FSAR 15.1.5), because the ANS Condition II criteria are met by the more challenging ANS Condition IV event.

II.2.5 Steam System Piping Failure - FSAR 15.1.5

This event was evaluated to assess the challenge to radiological dose criterion from fuel failure due to exceeding DNBR and/or fuel centerline melt limits. The methodologies used in the analyses of record were References II-7, II-8, and II-9. This event is analyzed at both HFP and HZP conditions.

HFP cases were initiated at the current licensed nominal power of 2900 MWt. HFP cases are driven by the maximum rate of positive moderator reactivity insertion, which is predominantly a function of the largest break flow rate (i.e., largest break size) and the most negative moderator temperature coefficient. The maximum break size and most negative moderator temperature coefficient are unchanged by the power uprate. The initial vessel average temperature also remains unchanged for the power uprate such that the extent of RCS cooldown is essentially the same as the analysis of record. Also, there will be no significant change in the positive Doppler reactivity feedback at uprate conditions, and the Doppler reactivity feedback is less significant than the large positive moderator reactivity feedback. The increase in power will not significantly change system response in the HFP analysis of record. The system response for the HZP cases is unaffected by the power uprate.

Since there is no significant change in the system response for the HFP cases at uprate conditions and the HZP cases are unaffected, the analysis of record is bounding for the MUR power uprate.

Radiological consequences of this event are discussed in Section II.2.38.

II.2.6 Loss of External Electrical Load - FSAR 15.2.2

This event is bounded by the Turbine Trip event (FSAR 15.2.3) per the methodology in Reference II-6.

II.2.7 Turbine Trip - FSAR 15.2.3

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.8 Inadvertent Closure of Main Steam Isolation Valves - FSAR 15.2.4

This event is bounded by the Turbine Trip event (FSAR 15.2.3) per the methodology in Reference II-6.

II.2.9 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip - FSAR 15.2.5

This event is bounded by the Turbine Trip event (FSAR 15.2.3) per the methodology in Reference II-6.

II.2.10 Loss of Non-Emergency AC Power to the Station Auxiliaries - FSAR 15.2.6

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

The radiological consequences of this event are discussed in Section II.2.38.

II.2.11 Loss of Normal Feedwater Flow - FSAR 15.2.7

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.12 Feedwater System Pipe Break - FSAR 15.2.8

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, and II-8. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

The radiological consequences of this event are bounded by the Steam System Piping Failure event (FSAR 15.1.5).

II.2.13 Partial Loss of Forced Reactor Coolant Flow - FSAR 15.3.1

This event is bounded by the Complete Loss of Forced Reactor Coolant Flow event (FSAR 15.3.2), because FSAR 15.3.2 meets the more restrictive ANS Condition II criteria.

II.2.14 Complete Loss of Forced Reactor Coolant Flow - FSAR 15.3.2

This event was evaluated to more restrictive ANS Condition II requirements. The analysis demonstrates that fuel failure will not occur. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8 and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.15 RCP Shaft Seizure (locked rotor) - FSAR 15.3.3

This event was evaluated to assess the challenge to radiological dose criterion from fuel failure due to penetration of the DNBR and/or fuel centerline melt limits. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, and II-8. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

The radiological consequences of this event are discussed in Section II.2.38.

II.2.16 RCP Pump Shaft Break - FSAR 15.3.4

This event is bounded by the RCP Shaft Seizure (locked rotor) event (FSAR 15.3.3) per the methodology in Reference II-6.

II.2.17 Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition - FSAR 15.4.1

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.18 Uncontrolled RCCA Bank Withdrawal at Power - FSAR 15.4.2

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. This event is evaluated at both 60% and HFP. The analyses of record assumed a maximum core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.19 Dropped Full Length RCCA or RCCA Bank - FSAR 15.4.3.1

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.20 Withdrawal of a Single Full Length RCCA - FSAR 15.4.3.2

This event was evaluated to assess the challenge to radiological dose criterion from fuel failure due to exceeding DNBR and/or fuel centerline melt limits. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8 and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

The radiological consequences of this event are discussed in Section II.2.38.

II.2.21 Statically Misaligned RCCA or Bank - FSAR 15.4.3.3

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The methodologies used in the analyses of record were References II-5, II-7, II-8, and II-10. The analyses are completed at nominal core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.22 Startup of an Inactive RCP at an Incorrect Temperature - FSAR 15.4.4

TS LCO 3.4.1.1 (Reactor Coolant System) prohibits power operation (Modes 1 & 2) with less than three reactor coolant loops in service. The consequences of the event in Modes 3-6 are bounded by the Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition event (FSAR 15.4.1). Therefore, this event is not analyzed. The start-up of an inactive loop event is not affected by the MUR power uprate.

II.2.23 CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant - FSAR 15.4.6

This event is analyzed primarily to assess the challenge to the time-to-criticality criteria. In Mode 1, the reactivity insertion rate resulting from this event is bounded by the range of reactivity insertion rates considered in the Uncontrolled RCCA Bank Withdrawal at Power (FSAR 15.4.2). The event is therefore bounded by the Uncontrolled RCCA Bank Withdrawal at Power event with respect to DNBR and fuel centerline melt criteria. Since the challenge to the acceptance criteria is most significant for operating modes other than Mode 1, the MUR power uprate related changes will not significantly impact this event.

II.2.24 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position - FSAR 15.4.7

This event was evaluated to assess the challenge to radiological dose criterion from fuel failure due to exceeding DNBR and/or fuel centerline melt limits. The methodologies used in the analyses of record were References II-5, II-7, and II-8. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

The radiological consequences of this event are discussed in Section II.2.38.

II.2.25 Spectrum of RCCA Ejection Accidents - FSAR 15.4.8

This event was evaluated to assess the challenge to radiological dose criterion from fuel failure due to exceeding DNBR and/or fuel centerline melt limits, primary system overpressure, and deposited fuel enthalpy. The methodologies used in the analyses of record were References II-6, II-7, II-8, II-10, and II-25. This event was evaluated at HZP and HFP initial conditions. HZP initial conditions are not impacted by the power uprate. For cases initiated at HFP conditions, the analyses assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

The radiological consequences of this event are discussed in Section II.2.38.

II.2.26 Inadvertent Operation of the ECCS During Power Operation - FSAR 15.5.1

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The challenge to primary system and secondary system overpressure criteria was also assessed, including confirmation that the pressurizer safety valve inlet conditions are acceptable. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.27 CVCS Malfunction that Increases Reactor Coolant Inventory - FSAR 15.5.2

An increase in reactor coolant system inventory resulting from the addition of cold, unborated water is analyzed in Section II.2.23 (CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant - FSAR 15.4.6). An increase in reactor coolant inventory resulting from injection of highly borated water is analyzed in Section II.2.26 (Inadvertent Operation of the ECCS During Power Operation - FSAR 15.5.1).

II.2.28 Inadvertent Opening of a Pressurizer Safety or PORV - FSAR 15.6.1

This event was evaluated to assess the challenge to the DNBR and fuel centerline melt criteria. The methodologies used in the analyses of record were References II-5, II-6, II-7, II-8, and II-10. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.29 Break in Instrument Line or Other Line from Reactor Coolant Pressure Boundary that Penetrate Containment - FSAR 15.6.2

The most severe radioactivity release from a failed line carrying primary coolant outside of containment is the rupture of the CVCS letdown line. The analyses of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

The analysis of the radiological consequences uses the analytical methods and assumptions outlined in SRP 15.6.2, since this accident is not discussed in RG 1.183. The offsite dose acceptance criteria from SRP 15.6.2 is designated as 10% of 10 CFR 100 limits. 10 CFR 100 has been replaced by 10 CFR 50.67. Applying this same basis to the 25 rem TEDE in 10 CFR 50.67, the offsite dose limit is 2.5 rem TEDE. The radiological criterion for the control room dose is 5.0 rem TEDE per 10 CFR 50.67. The resulting doses to the EAB, LPZ and control room from this event are within the acceptance criteria. Therefore the current analysis remains bounding at MUR power uprate conditions.

II.2.30 Steam Generator Tube Rupture - FSAR 15.6.3

The analysis for the SGTR event is performed to demonstrate that the offsite radiological consequences remain below the guideline values and that SG overfill does not occur. The thermal-hydraulic analysis of record uses the NRC approved LOFTTR2 methodology (References II-11 and II-12) to calculate the ruptured SG primary-to-secondary break flow and steam released to the environment. In this analysis, the primary-to-secondary break flow and the steam releases to the atmosphere from both the ruptured and intact SGs were calculated for use in determining the activity released to the atmosphere. The mass releases were calculated from the event initiation until break flow termination. The mass release information is used to calculate the radiation doses at the EAB, LPZ, and control room. In addition to the thermal-hydraulic analysis, a margin to SG overfill analysis was performed. The analysis concluded that SG overfill does not occur. These analyses assumed a core power of 102% of NSSS power 2912.4 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

The radiological consequences of this event are discussed in Section II.2.38.

II.2.31 Large Break Loss of Coolant Accident - FSAR 15.6.5.2

This event is analyzed to access the challenge to the 10 CFR 50.46 criteria, particularly peak clad temperature and oxidation limits. The LBLOCA methodology is currently EMF-2087(P)(A) (Reference II-26). The Appendix K analysis of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

An LAR (Reference II-38) was submitted to adopt the AREVA Realistic LBLOCA methodology, EMF-2103(P)(A) (Reference II-45) and add this methodology to the TS. The NRC has previously approved the EMF-2103(P)(A) methodology (Reference II-39). However, the LAR was subsequently withdrawn to supplement the analysis to address generic NRC concerns with using EMF-2103(P)(A). A revised submittal will contain plant specific methodology that takes exception to EMF-2103(P)(A) in those areas where the NRC staff identified specific concerns during their review of Reference II-38. The LBLOCA methodology will use a safety analysis core power of 2958 MWt and thus be neutral relative to the MUR power uprate LAR review. Both analyses bound the proposed MUR power uprate.

The radiological consequences of this event are discussed in Section II.2.38.

II.2.32 Small Break Loss of Coolant Accident - FSAR 15.6.5.3

This event is analyzed to access the challenge to the 10 CFR 50.46 criteria, particularly peak clad temperature and oxidation limits. The SBLOCA was analyzed using the methodologies in References II-27 and II-28. The Appendix K analysis of record assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

A pending LAR (Reference II-42) regarding M5™ cladding will adopt the EMF-2328(P)(A) (Reference II-44) methodology and supersede XN-NF-82-49 for SBLOCA. EMF-2328(P)(A) is currently approved for use at HNP and is specified in TS 6.9.1.6.2.m, but has not been used as the analysis of record for SBLOCA. After the transition to M5™ cladding, XN-NF-82-49 will no longer be applicable to HNP. The SBLOCA safety analysis power level remains at 2958 MWt, or 102% of 2900 MWt.

The radiological consequences of this event are discussed in Section II.2.38.

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II.2.33 Radioactive Waste Gas System Leak or Failure - FSAR 15.7.1

The limiting event is the failure of a single gaseous waste decay tank. Tank inventory is based on RG 1.24 assessment of maximum activity associated with post-shutdown degassing, with 1% fuel cladding defects. The analysis is performed using the AST methodology described in RG 1.183. The offsite dose acceptance criterion for a gas decay tank rupture is defined in HNP TS 6.8.4.j and RG 1.183 as 0.5 rem whole body. This correlates to a dose limit of 0.5 rem TEDE. The control room dose limit is 5.0 rem TEDE per 10 CFR 50.67. The gas decay tank rupture dose to the EAB, LPZ and control room are below the acceptance criteria. This analysis assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.34 Liquid Waste System Leak or Failure - FSAR 15.7.2

The postulated doses from this class of events have been historically small. This accident analysis has been deleted, consistent with the deletion of SRP 15.7.2 guidance. The liquid radioactive water system failure is bounded by the Postulated Radioactive Releases Due to Liquid Tank Failure FSAR 15.7.3.

II.2.35 Postulated Radioactive Releases Due to Liquid Tank Failure - FSAR 15.7.3

Analysis of potential releases from a RWST failure assumes 2958 MWt or 102% of 2900 MWt with 1% failed fuel. The analysis indicated that ground and surface water transport pathways are such that the results are within with the maximum permissible concentrations limits as established in the 10 CFR 20 version used for the original plant license. The current analysis remains bounding for the MUR power uprate.

II.2.36 Design Basis Fuel Handling Accidents - FSAR 15.7.4

The current fuel handling accident radiological analysis is based upon the AST, with acceptance criteria as specified in either 10 CFR 50.67 or RG 1.183. The existing fuel handling accident dose evaluation was performed using a core inventory that assumes 2958 MWt or 102% of 2900 MWt. Two cases were considered: accident in the containment and accident in the FHB.

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II.2.36.a Containment Fuel Handling Accident

The fuel handling accident in containment analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly. Thus, the analysis supports the design basis limit of 100 hours decay time prior to fuel movement. The isotopes released from one fuel assembly (264 fuel rods) are assumed to enter the water and move into the building's atmosphere. The offsite dose limit is 6.3 rem TEDE per RG 1.183. This is approximately 25% of the 10 CFR 50.67 guideline. The limit for the control room dose is 5.0 rem TEDE per 10 CFR 50.67. The doses to the EAB, LPZ, and control room are within the established limits. The current analysis remains bounding at MUR power uprate conditions.

II.2.36.b FHB Fuel Handling Accident

This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly on top of another recently discharged PWR fuel assembly in a fuel storage rack. Thus, the analysis supports the design basis limit of 100 hours decay time prior to fuel movement. The dropped fuel assembly subsequently falls over landing on BWR fuel assemblies in an adjacent storage rack. Fifty fuel rods are projected to fail in the impacted PWR fuel assembly in storage, and all of the fuel rods (264) in the dropped fuel assembly fail when the assembly falls over. Due to the upper bail handle of the BWR fuel assemblies extending above the top of the BWR storage racks, up to 52 BWR fuel assemblies could be impacted when the dropped PWR fuel assembly falls over. All of the fuel rods in the impacted BWR assemblies are assumed to fail. The BWR fuel assemblies are four year decayed Brunswick fuel assemblies. The offsite dose limit is 6.3 rem TEDE per RG 1.183. This is approximately 25% of the 10 CFR 50.67 guideline. The limit for the control room dose is 5.0 rem TEDE per 10 CFR 50.67. The doses to the EAB, LPZ, and control room are within the established limits. The current analysis remains bounding at MUR power uprate conditions.

II.2.37 Anticipated Transient Without Scram - FSAR 15.8

An ATWS event is defined as an anticipated operational occurrence (such as a loss of normal feedwater, loss of condenser vacuum, or LOOP) combined with an assumed failure of the reactor trip system to shutdown the reactor. For PWRs manufactured by Westinghouse, the ATWS rule requirements are specified in 10 CFR 50.62, *Requirements for Reduction of Risk from Anticipated Transient Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plant*. HNP satisfied 10 CFR 50.62 by installing the NRC approved AMSAC. The AMSAC system automatically initiates a turbine trip and starts auxiliary feedwater pumps under conditions indicative of an ATWS and a loss of main feedwater. AMSAC was described in an NRC submittal (Reference II-29), which confirmed that the Westinghouse generic analyses in Reference II-30 were applicable to HNP. The NRC approved the HNP AMSAC design in Reference II-14.

The AMSAC system and the analyses were reviewed with respect to the proposed MUR power uprate. HNP is a 3-loop PWR with Model Delta-75 replacement SGs. The Model 44 SG ATWS model and analysis were modified for the Delta-75 replacement SGs. The two most limiting RCS overpressure transients from the Westinghouse generic analyses are the loss of normal feedwater and loss of load. An analysis was performed using the LOFTRAN computer code to assess the effect of the MUR power power on the most limiting transients. The peak RCS pressure at uprate conditions is less than the ASME B&PV Code Service Level C acceptance criterion of 3200 psig. The evaluation demonstrated that the AMSAC system will continue to meet the requirements of 10 CFR 50.62 at MUR power uprate conditions.

The AMSAC design specifies a nominal permissive C-20 arming setpoint based on the generic 40% setpoint, minus an allowance for inaccuracies in the turbine first stage impulse pressure channels. The turbine first stage impulse pressure channels will require scaling changes due to the increase in turbine first stage impulse pressure at the uprated full power level. There are no other AMSAC impacts as a result of the MUR power uprate.

II.2.38 Radiological Consequences

The radiological consequences analyses in the FSAR have been evaluated for the power uprate. The dose analyses are potentially impacted in three areas: the core and coolant activities prior to the accident (source term), the fuel failure resulting from the accident, and the secondary side steam releases following the accident. The analyses are based upon the AST, with acceptance criteria as specified in either 10 CFR 50.67 or RG 1.183. The radiological consequences analyses were performed using the core inventory that assumes 2958 MWt or 102% of 2900 MWt, and therefore remain applicable at power uprate conditions. The fuel failure and melting assumptions are not changing and will be verified during the standard core reload process. The steam release rates used in the following accident analyses have been recalculated to reflect the change in power measurement uncertainty: MSLB, loss of AC (offsite) power, locked rotor, single RCCA withdrawal, rod ejection, and SBLOCA. Revised steam releases and feedwater flows were calculated at 102% of 2900 MWt plus 12.4 MWt for RCP heat, and bound the uprated power level including uncertainty. The standard steam release for dose calculation analyzes four steam release events: steam line break, locked rotor, loss of AC (offsite) power and loss of load. The loss of load event is not part of the HNP radiological licensing basis, so the only radiological accidents included in the steam mass release analysis are the steam line break, locked rotor and loss of AC (offsite) power. Table II-2 provides the steam releases from the intact-loop SGs and main feedwater flows for the 0-2 hour and 2-8 hour time periods for the steam line break, locked rotor, and loss of AC (offsite) power events. For the steam line break event, additional steam is released through the faulted-loop SG from the transient initiation through assumed main and auxiliary feedwater flow termination. The steam mass released from the faulted-loop SG is 162,000 lbm.

Table II-2
Steam Released and Main Feedwater Flows

Event	Steam Release (lbm)		Feedwater Flow (lbm)	
	0-2 hours	2-8 hours	0-2 hours	2-8 hours
Steam Line Break	401,000 (MUR) 386,000 (AOR)	917,000 (MUR) 892,000 (AOR)	494,000 (MUR) 482,000 (AOR)	983,000 (MUR) 967,000 (AOR)
Locked Rotor, Loss of Offsite AC Power	378,000 (MUR) 364,000 (AOR)	965,000 (MUR) 939,000 (AOR)	517,000 (MUR) 508,000 (AOR)	1,064,000 (MUR) 1,052,000 (AOR)

The revised steam release rates and feedwater flows were used in the radiological dose analysis to determine the revised doses for the MSLB, loss of AC (offsite) power, locked rotor, single RCCA withdrawal, rod ejection, and SBLOCA analyses. The revised doses are shown in Table II-3 with the associated dose acceptance limits summarized in Table II-4.

Table II-3
Revised Accident Doses

Event	Doses (rem TEDE)		
	EAB	LPZ	Control Room
Main Steam Line Break (Pre-Accident Iodine Spike)	0.14	0.15	0.38
Main Steam Line Break (Accident Initiated Iodine Spike)	0.73	1.09	2.58
Main Steam Line Break (Fuel Failure)	1.51	2.62	4.11
Loss of AC (offsite) Power (Pre-Accident Iodine Spike)	0.013	0.0096	0.029
Loss of AC (offsite) Power (Accident Initiated Iodine Spike)	0.045	0.023	0.069
Locked Rotor	1.97	1.46	3.31
Single RCCA Withdrawal	1.64	1.28	2.74
Rod Ejection	4.04	4.13	4.60
SBLOCA	9.57	3.70	4.92

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Table II-4
Dose Acceptance Limits

Event	Dose Limits [1] (rem TEDE)		
	EAB	**LPZ**	**Control Room**
Main Steam Line Break (Pre-Accident Iodine Spike)	25	25	5
Main Steam Line Break (Accident Initiated Iodine Spike)	2.5	2.5	5
Main Steam Line Break (Fuel Failure)	25	25	5
Loss of AC (offsite) Power (Pre-Accident Iodine Spike)	25	25	5
Loss of AC (offsite) Power (Accident Initiated Iodine Spike)	2.5	2.5	5
Locked Rotor	2.5	2.5	5
Single RCCA Withdrawal	2.5	2.5	5
Rod Ejection	6.3	6.3	5
SBLOCA	25	25	5

1. These criteria have not changed for the MUR power uprate.

The release pathways, X/Qs, and dose conversion factors are unchanged from the AST license amendment request and associated amendment (References II-18 and II-13). The accident doses were evaluated considering source terms, the fuel failure and melting assumptions, and the post-trip steam releases. These areas are unaffected except for the steam releases, which were evaluated by increasing the affected doses. The accident doses at MUR power uprate conditions remain below the applicable dose acceptance limits.

II.2.38.a Steam Generator Tube Rupture Dose Evaluation

The current SGTR radiological analysis is based upon the AST as defined by NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or RG 1.183. The analysis involves the complete severance of a single SG tube. Due to the pressure differential between the primary and secondary systems, primary coolant is released to the secondary side of the SG and then to the environment. The source terms for equilibrium conditions with 1% failed fuel are normalized to the TS Dose Equivalent Iodine 131 limits in the primary coolant. Thus, the distribution of iodine isotopes is not sensitive to small changes in core power and using iodine concentrations based on the uprated power level has no impact on the analysis results. The site boundary and LPZ offsite dose limit for a SGTR with an assumed pre-accident iodine spike is the RG 1.183 specified 25 rem TEDE. The site boundary and LPZ offsite dose limit with an assumed accident initiated iodine spike is the RG 1.183 specified 2.5 rem TEDE. The control room dose limit for both the assumed pre-accident iodine spike and the assumed accident initiated iodine spike is 5.0

rem TEDE per 10 CFR 50.67. The doses to the site boundary, LPZ, and control room are within the established limits. This analysis assumed a core power of 2958 MWt or 102% of 2900 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.38.b Large Break LOCA Evaluation

The current LOCA radiological analysis is based upon the AST as defined by NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or RG 1.183. The existing TS 5.3.1 restricts fuel enrichment to 5.0 w/o U-235, which is unchanged by the power uprate. Fuel assembly exposure is restricted to AREVA's peak rod average burnup limit of 62,000 MWD/MTU. The power uprate results in limited changes to core power and burnup history. The offsite dose limit is the RG 1.183 specified 25 rem TEDE. This is the guideline value of 10 CFR 50.67. The control room dose limit is 5.0 rem TEDE per 10 CFR 50.67. The doses to the EAB, LPZ, and control room are within the established limits. The current LOCA dose analysis is based on a core inventory that assumes 2958 MWt or 102% of 2900 MWt. Therefore, the existing LOCA radiological analysis remains bounding for the MUR power uprate.

II.2.38.c CVCS Letdown Line Break Evaluation

The CVCS letdown line break evaluation is discussed in Section II.2.29.

II.2.38.d Radioactive Waste Gas System Leak or Failure Dose Evaluation

The radioactive waste gas system leak or rupture dose evaluation is discussed in Section II.2.33.

II.2.38.e Liquid Waste System Leak or Failure Dose Evaluation

The liquid waste system leak or failure dose evaluation is discussed in Section II.2.34.

II.2.38.f Postulated Radioactive Releases Due to Liquid Tank Failure

The radioactive liquid tank failure dose evaluation is discussed in Section II.2.35.

II.2.38.g Fuel Handling Accident Dose Evaluation

The fuel handling accident dose evaluation is discussed in Section II.2.36.

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II.2.39 Natural Circulation - FSAR 15.2.6

HNP satisfied the functional requirements of the natural circulation cooldown through a comparison of test results from a previously tested plant of similar design, along with a supporting thermal-hydraulic analysis of a plant-specific cold shutdown scenario. The analysis of record is a comparison and evaluation of the results from the Diablo Canyon Natural Circulation/Boron Mixing/Cooldown Test.

An evaluation was performed to confirm the natural cooldown capability at power uprate conditions and determine if the existing analysis of record was bounding. This analysis compared relevant plant parameters to the natural circulation cooldown test performed by Diablo Canyon Unit 1. The reactor core power level was conservatively assumed at 2962 MWt in the analysis of record. This power level bounds the MUR uprate power level. Other important parameters including T_{avg}, feedwater temperature, SG tube plugging and thermal design flow are also bounded by the analysis of record. The previous analysis conclusions remain valid at power uprate conditions.

II.2.40 Mass and Energy Releases - FSAR 6.2.1

The method used to calculate the LOCA mass and energy release for FSAR Section 6.2.1 is an NRC approved method (Reference II-16). Westinghouse informed PEC that it had discovered several generic issues that affect the LOCA long-term mass and energy releases. These issues were identified independently of the HNP MUR power uprate. One of these issues is an EPITOME code error. EPITOME is a code used by Westinghouse in the LOCA long-term mass and energy calculation, which supplies input to the containment response. This issue applies generically to the users of this Westinghouse methodology. The HNP power uprate LOCA long-term mass and energy evaluation was predicated on no changes to the existing mass and energy release analysis of record due to the power uprate. That evaluation remains applicable; that is the MUR power uprate conditions do not cause an increase in the LOCA mass and energy releases. However, the recently identified generic issues affect the analysis of record values. PEC is aggressively working to reconcile the analysis of record impact due to these generic issues. The short-term LOCA mass and energy used for subcompartment analyses and the main steam line break mass and energy release are not impacted by the identified generic issues.

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II.2.40.a Long-term LOCA Mass and Energy Release Analysis

The long-term LOCA mass and energy releases used in the FSAR Chapter 6 containment analyses were submitted to the NRC in Reference II-16. This evaluation model has been reviewed and approved generically by the NRC. The approval letter is included with Reference II-16. The mass and energy release evaluation model is comprised of the following codes: SATAN, WREFLOOD, FROTH, and EPITOME. These codes have been used for this analysis since the original plant licensing. The analyses assumed a core power of 2958 MWt or 102% of 2900 MWt.

In 2005, deficiencies were identified in the current LOCA mass and energy analysis that resulted in three long-term penalties. These penalties were: the assumed area of the upper plenum, addition of main feedwater post-LOCA, and purge of the main feedwater lines by the auxiliary feedwater flow. The auxiliary feedwater purge penalty does not apply to HNP because there are dedicated lines to the SGs metal mass that is above the operating level. The other penalties are long-term and do not apply to the blowdown limited double-ended hot leg break, which is the limiting break at HNP. The upper plenum area error and main feedwater addition long-term penalties apply to the double-ended pump suction break. However, credit for inactive metal heat energy in the SG secondary is available and offsets these two long-term penalties. Therefore, the current long-term LOCA mass and energy analyses are unaffected by the MUR power uprate.

II.2.40.b Short-term LOCA Mass and Energy Release Analysis

FSAR Section 6.2.1.2 describes the analyses of containment subcompartment response post-LOCA. Containment subcompartments are subject to pressure transients and jet impingement forces caused by the mass and energy releases from postulated high energy pipe ruptures within their boundaries. Analysis was performed to ensure that the subcompartment walls can maintain structural integrity during the short pressure pulse (generally less than three seconds) accompanying a high energy line pipe rupture within that subcompartment. Subcompartments where high energy ruptures are postulated include the reactor cavity, pressurizer subcompartment, and the three SG subcompartments. The methodology described in Reference II-31 was used for these analyses. The NRC has determined that the Westinghouse mass and energy models described in Reference II-31 satisfy the expectations of NUREG-0800, Section 6.2.1.3, Subsection II regarding LOCA mass and energy release calculations.

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The original HNP design and licensing bases did not include leak-before-break methodology. As a result, the dynamic effects of large RCS pipe breaks were also considered in the structural design basis for the containment. The following short-term LOCA mass and energy releases were originally considered for HNP as indicated in FSAR Section 6.2.1.2:

Case 1	150 in^2 cold leg break (reactor cavity blowdown)
Case 2	150 in^2 hot leg break (reactor cavity blowdown)
Case 3	Double-ended cold leg break
Case 4	Double-ended hot leg break
Case 5	Double-ended pump suction break
Case 6	Double-ended pressurizer surge line break
Case 7	Pressurizer spray line break

The short-term LOCA mass and energy releases are affected by reductions in RCS temperatures, due to the fluid density effect on the initial pressure pulse created when the pipe ruptures. HNP is approved for leak-before-break, so Case 1 through 5 breaks have been eliminated and only breaks in the largest branch lines (Cases 6 and 7) require evaluation. The power uprate impact on the current short-term LOCA licensing basis was evaluated. This evaluation assumed a vessel outlet temperature (T_{hot}) of 602°F and a vessel/core inlet temperature (T_{cold}) of 530°F. The pressurizer surge line break and pressurizer spray line break analyses were based on a T_{hot} of 601.4°F and a T_{cold} of 548.4°F. Since the RCS T_{hot} increased for the power uprate, there is no impact on the pressurizer surge line break (Case 6). However, the RCS T_{cold} value decreased by 18.4°F, which results in a 4.7% increase in the pressurizer spray line mass and energy release (Case 7).

The power uprate effect on the current licensing basis was qualitatively evaluated by comparing the postulated line break mass and energy releases at uprate conditions to the current conditions. Since RCS piping breaks have been eliminated by the leak-before-break methodology, and previous assessments indicated that the RHR and accumulator line breaks near the reactor cavity and in the SG subcompartments are bounded by the original analyses, the only breaks evaluated for the power uprate are those in the pressurizer subcompartment (pressurizer surge line and pressurizer spray line breaks). Power uprate conditions do not affect the pressurizer surge line mass and energy releases, because the RCS hot leg temperature increases. The 4.7% increase in the pressurizer spray line mass and energy release is small and the pressurizer spray line is not limiting for the pressurizer subcompartment design basis. Therefore, the current subcompartment analyses are unaffected by the MUR power uprate and remain bounding.

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II.2.40.c Main Steam Line Break and Feedwater Line Break Mass and Energy Release

The long-term MSLB mass and energy releases used in the FSAR Chapter 6 containment analyses were analyzed with the LOFTRAN computer code (Reference II-32). The use of the LOFTRAN code is documented in WCAP-8822, Supplement 1 (Reference II-33), which has been reviewed and approved by the NRC for this application. The analyses of record for long-term steamline breaks inside and outside containment assumed a core power of 102% of 2900 MWt with the addition of 12.4 MWt for RCP heat.

There is no effect on either the current licensing basis for the long-term steamline break mass and energy release analysis or the FSAR conclusions as a result of the MUR power uprate. The only critical parameter for the short-term feedwater line break is the maximum SG pressure. The bounding value for the SG pressure remains valid at uprate conditions. Therefore, the mass and energy releases remain appropriate.

II.2.41 Station Blackout - FSAR 8.3.1.2.21

Station blackout (SBO) is discussed in Section V.1.B.

II.2.42 Analysis to Determine EQ Parameters - FSAR 3.11

The following conditions related to EQ were evaluated: LOCA and MSLB pressure/temperature inside containment, MSLB/main feedwater line break pressure/temperature outside containment, other HELBs pressure/temperature outside containment, and radiation effects both normal operation and accident.

The pressure/temperature profiles calculated for both a LOCA and MSLB inside containment were performed using a power level of 102% of rated power. The mass and energy release calculations were performed assuming a power level of 102% of rated power. There are no significant changes to the analysis inputs and the current analysis of record remains bounding for the power uprate. For the MSLB/main feedwater line break outside containment, the worst case condition affecting EQ was in the main steam tunnel. These analyses were performed using a power level of 102% of rated power. The current analysis of record remains bounding for the power uprate. Other HELBs outside containment were determined to either have no essential components located in the vicinity of the break, or the effects of the break would be mitigated by the large volume of the area where the break occurred. The current analysis of record remains

bounding for the power uprate. The existing analysis of record regarding accident and normal operation dose assumed a power level of 102% of rated power. Thus, the proposed power uprate will not affect the accident dose estimates inside and outside containment, and will continue to be conservative for the normal operation dose estimates in the RAB. Due to available margin resulting from the current methodology, the existing normal operation dose estimates for inside containment locations remain applicable for the power uprate.
Therefore, the current analyses of record remain bounding and the MUR power uprate will not adversely affect equipment in the EQ Program for environmental qualification.

EQ of electrical equipment is discussed in Section V.1.C.

II.2.43 Safe Shutdown Fire Analysis - FSAR 9.5.1

FSAR Section 9.5.1 describes the fire protection system and associated design bases. HNP has transitioned from the former deterministic fire protection program and licensing basis to a risk-informed, performance-based fire protection program as described in 10 CFR 50.48(c). 10 CFR 50.48(c) endorses, with exceptions, NFPA 805-2001 (Reference II-23). NFPA 805 is a national consensus standard that allows plants to use engineering analyses to demonstrate that the installed fire protection systems and features are sufficient to meet specific fire protection and nuclear safety goals, objectives and performance criteria. It allows the use of performance-based methods such as fire modeling, and risk-informed methods such as fire probabilistic risk assessment, to demonstrate compliance with these performance criteria. The nuclear safety performance criteria include provisions for ensuring that reactor reactivity control, RCS inventory and pressure control, decay heat removal, vital auxiliaries, and process monitoring are achieved and maintained. HNP has adopted NFPA 805-2001 through a license amendment (Reference II-22). HNP also uses the guidance in NEI 04-02 (Reference II-34) as endorsed by RG 1.205 (Reference II-35). NEI 04-02 provides guidance on the overall process (programmatic, technical, and licensing) of transitioning from the traditional deterministic fire protection licensing basis to a new one based upon NFPA 805-2001.

The Safe Shutdown Analysis supporting calculations provide the basis, methodology and results to ensure that the ability to achieve the nuclear performance criteria of NFPA 805-2001 is maintained for a postulated fire at HNP. Reviews were conducted to evaluate the power uprate impact on the existing Safe Shutdown Fire Analysis. The power uprate will not result in functional changes to any system pertaining to safe shutdown. New components or cables are not added to any fire safe shutdown systems, so safe shutdown separation is not impacted. HNP maintains the capability to achieve cold shutdown (reactor core at 0% thermal power excluding decay heat, k_{eff} less than 0.99, and RCS average temperature less than 200°F) within the 72 hour limit. The power uprate does not affect the present combustible loading. Operator actions are discussed in Section VII.1.

The review identified one supporting calculation (HNP-F/NFSA-0171, Reference II-36) that was completed at 2900 MWt and required updating to 2958 MWt (102% of 2900 MWt). Calculation HNP-F/NFSA-0171 evaluated the RCS cooldown without boration event for the power uprate. During this event, emergency boration capabilities are incapacitated by a fire, and the only source for boric acid additions is the RWST. The updated calculation concluded that the power uprate does not preclude safe shutdown for this event. The remaining supporting calculations were unaffected by the power uprate. Therefore, the safe shutdown fire analyses remain applicable at MUR power uprate conditions.

II.2.44 Spent Fuel Pool Cooling - FSAR 9.1.3

The fuel handling building is split into two storage facilities. The storage facility on the South end consists of two SFPs (Pool A and B). The storage facility on the North end consists of two SFPs (Pool C and D). The SFPs are designed to accommodate both new and spent fuel.

FSAR Section 9.1.3 describes the cooling requirements for the SFP. The analysis considers Normal Operation, Incore Shuffle, Normal Full Core Offload, and Abnormal Core Offload cases. Each scenario assumes that fuel movement begins no earlier than the limits provided in the Relocated TS and Design Basis Requirements procedure PLP-114 (Reference II-40) for movement of irradiated fuel. Calculations of the maximum thermal energy removed by the SFP cooling system were originally conducted using the POOLHEAT code, with an assumption of a maximum 2900 MWt core power level. HNP currently uses the ORIGEN2 code, which assumes a core thermal power of 2900 MWt and adds a 2.0% reactor power uncertainty. The POOLHEAT and ORIGEN2 results were compared for the design heat loads on the fuel pool cooling and cleanup system in calculation HNP-F/NFSA-0071 (Reference II-37). For SFP A and B, the POOLHEAT results bound the ORIGEN 2 results for Normal Operation, Incore Shuffle, Normal Full Core Offload, and Abnormal Core Offload conditions. The heat load for SFP C and D is limited to 7.0 Mbtu/hr by TS 5.6.3d. This limit is not affected by the power uprate. There is currently no HNP fuel stored in SFP C and D.

Spent fuel pool bounding heat loads and fuel pool cooling system performance parameters are not affected by the power uprate. Therefore, the SFP cooling system has the capability to maintain SFP temperature within the existing 150°F limit at MUR power uprate conditions.

Refer to Section VI.1.D for further discussion on the spent fuel pool storage and cooling.

II.2.45 Internal Flooding - FSAR 3.6A.3.2 and 3.6A.6

The design bases for flooding inside and outside the reactor containment building were evaluated.

The maximum flood level inside the reactor containment building is at the 228.6 ft. elevation, based on the post-LOCA containment flood level calculation performed at 2958 MWt or 102% of 2900 MWt. All safety-related equipment in the reactor containment building is located above this calculated water level. Therefore, no flooding analysis is required. The power uprate does not affect the existing post-LOCA containment flood level calculation, so the existing flooding protection is adequate for power uprate conditions.

The bounding flood conditions in areas outside containment occur as a result of HELBs and MELBs. The single-ended rupture of the largest feedwater line is the bounding flooding case due to HELBs/MELBs in the main steam and feedwater tunnel. All equipment in this area that is required to achieve and maintain safe plant shutdown is located above the calculated flood level. Outside of the main steam and feedwater tunnel, the worst case flooding in the RAB results from postulated cracks in moderate energy piping. The analysis concluded that the maximum flood level resulting from MELBs in the RAB was one foot above the building floor. All safety-related equipment in this building is located above the calculated one foot elevation or is environmentally qualified for its environmental condition, including submergence.

Flooding analyses for the turbine and fuel handling buildings are not required because there is no equipment located in these buildings essential for safe plant shutdown.

Based on flooding calculation reviews, it was determined that the MUR power uprate does not affect the existing postulated flooding conditions inside or outside the reactor containment building.

II.3 Design Transients

II.3.1 Nuclear Steam Supply System Design Transients

NSSS design transients were specified in the original design analyses of NSSS components cyclic behavior. The selected transients are conservative representations of transients that when used as a basis for component fatigue analysis, provide confidence that the component is appropriate for its application over the 60-year plant license period. The RCS and its auxiliary system components are designed to withstand the cyclic load effects from RCS temperature and pressure changes. The existing design transients were evaluated for their continued applicability at MUR power uprate conditions.

The key plant design parameters for the NSSS design transients are RCS hot and cold leg temperatures (T_{hot}, T_{cold}), secondary side steam temperature and pressure (T_{steam}, P_{steam}), and the secondary side feedwater temperature. The existing design transients for parameters bound plant operation at the uprated conditions. The component fatigue evaluation results are discussed in Section IV.

The frequencies of occurrence for the 60-year plant licensed period are unchanged and new design transients are not created as a result of the MUR power uprate.

II.3.2 Auxiliary Equipment Design Transients

The auxiliary equipment design specifications included transients that were used to design and analyze the Class 1 auxiliary nozzles connected to the RCS, and certain NSSS auxiliary systems piping, heat exchangers, pumps and tanks. The transients are sufficiently conservative, such that when used as a basis for component fatigue analysis, they provide confidence that the component will perform as intended over the plant operating license period.

The only auxiliary equipment design transients potentially impacted by the power uprate are those transients associated with full load NSSS design temperatures (T_{hot} and T_{cold}). These temperature transients are defined by the differences between RCS loop coolant temperature and the temperature of coolant in the auxiliary systems connected to the RCS loops. Since the operating coolant temperatures in the auxiliary systems are not impacted by the power uprate, the temperature difference between auxiliary systems and the RCS loops is only affected by changes in the RCS operating temperatures. The transients assume a full load NSSS T_{hot} and T_{cold} of

630°F and 560°F, respectively. These full load temperatures were selected for equipment design to ensure that the temperature transients would be conservative for a wide range of NSSS design parameters. The approved NSSS design temperature ranges for T_{hot} and T_{cold} used to develop the current design transients are higher than the power uprate temperatures. Less severe transients result from the lower full load temperatures at power uprate conditions. Therefore, the existing auxiliary equipment design transients are conservative and bounding for the MUR power uprate.

II.3.3 NSSS Pressure Control Component Sizing

The pressure control component sizing was evaluated at power uprate conditions. RCS pressure control component sizing includes the pressurizer PORVs, spray valves, and heater capacities. These components must continue to successfully perform their intended functions.

- The pressurizer PORV sizing basis prevents the pressurizer pressure from reaching the high pressure reactor trip setpoint for the design basis load rejection with steam dump transient. The PORVs would not be challenged for a 50% load rejection. The installed PORV capacity remains acceptable at uprate conditions.
- The pressurizer spray capacity sizing basis is to withstand a 10% step load decrease transient without pressurizer PORV actuation. The limiting transient analysis demonstrates that the design pressurizer spray capacity is adequate to maintain pressurizer pressure below the PORV actuation setpoint. The installed pressurizer spray capacity remains acceptable at uprate conditions.
- The pressurizer heaters are sized to prevent actuating the low pressurizer pressure reactor trip setpoint (for Condition I transients), or the low pressurizer pressure safety injection setpoint (for a reactor trip transient). The design basis of one kilowatt per one cubic foot of pressurizer volume is met at uprate conditions. The required heating capacity and heat up time from cold shutdown to hot standby are not impacted by the power uprate. The installed pressurizer heater capacity remains acceptable at uprate conditions

The existing pressure control components (PORVs, spray valves, and heaters) meet the sizing criteria at the MUR power uprate conditions. The component capacities are adequate to mitigate the sizing basis transients without exceeding the limits.

II.3.4 Plant Operability Margin to Trip

The plant operability margin to trip was evaluated to ensure that the operating margins to relevant reactor protection system and ESF actuation system trip setpoints are adequate during ANS Condition I (normal condition) transients at power uprate conditions. Plant operability for ANS Condition I transients includes plant response to 5% per minute loading and unloading, 10% step load increase or decrease, large (50%) load rejection, and turbine trip followed by a reactor trip from full power.

Most of the analyses of record assumed a 2.0% power uncertainty for conservatism. These operability analyses are best estimate, so the 2.0% assumed uncertainty bounds the power uprate. The NSSS control systems design, setpoints and time constants are consistent with the analysis of record. The fuel assumed in the analysis of record remains unchanged for the power uprate.

The evaluation concluded that adequate margin exists to relevant reactor trip and ESFAS setpoints during the normal condition transients at uprated power conditions. The NSSS control systems provide a stable and acceptable response to normal condition transients at uprated power conditions.

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III. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

1. This section covers the transient and accident analyses that are included in the plant's UFSAR (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of

environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling and flooding).

2. For analyses that are covered by the NRC approved reload methodology for the plant, the licensee should:

    A. Identify the transient/accident that is the subject of the analysis
    B. Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate

    C. Provide an explicit commitment to submit the analysis for NRC review, prior to operation at the uprated power level, if NRC review is deemed necessary by the criteria in 10 CFR 50.59

    D. Provide a reference to the NRC's approval of the plant's reload methodology

3. For analyses that are not covered by the reload methodology for the plant, the licensee should provide a detailed discussion for each analysis. The discussion should include:

    A. Identify the transient or accident that is the subject of the analysis

    B. Identify the important analysis inputs and assumptions (including their values), and explicitly identify those that changed as a result of the power uprate

    C. Confirm that the limiting event determination is still valid for the transient or accident being analyzed

    D. Identify the methodologies used to perform the analyses, and describe any changes in those methodologies

    E. Provide references to staff approvals of the methodologies in Item D. above

    F. Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology

G. Describe the sequence of events and explicitly identify those that would change as a result of the power uprate

H. Describe and justify the chosen single-failure assumption

I. Provide plots of important parameters and explicitly identify those that would change as a result of the power uprate

J. Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head, and valve isolation capabilities) required to support the analysis

K. Discuss the results and acceptance criteria for the analysis, including any changes from previous analysis

RESPONSE TO III. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

There are no accidents or transients where the existing analyses of record do not bound plant operation at the proposed uprated power level.

IV. MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

1. A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified in Section II, above. For components that are not bounded by existing analysis of record, a detailed discussion should be provided.

   A. This discussion should address the following components:

      i. reactor vessel, nozzles and supports

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ii. reactor core support structures and vessel internals
iii. control rod drive mechanisms
iv. Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles
v. balance-of-plant (BOP) piping (NSSS interface systems, safety related cooling water systems, containment systems)
vi. steam generator tubes, secondary side internal support structures, shell, nozzles
vii. reactor coolant pumps
viii. pressurizer shell, nozzles, surge line
ix. safety-related valves

B. The discussion should identify and evaluate any changes related to the power uprate in the following areas:

i. stresses
ii. cumulative usage factors (fatigue)
iii. flow induced vibration
iv. changes in temperature (pre- and post-uprate)
v. changes in pressure (pre- and post-uprate)
vi. changes in flow rates (pre- and post-uprate)
vii. high energy line break locations
viii. jet impingement and thrust forces

C. The discussion should also identify any effects of the power uprate on the integrity of the reactor vessel with respect to:

i. pressurized thermal shock calculations
ii. fluence evaluation
iii. heatup and cooldown pressure-temperature limit curves
iv. low temperature overpressure protection
v. upper shelf energy
vi. surveillance capsule withdrawal schedule

D. The discussion should identify the code of record being used in the associated analyses, and any changes to the code of record.

E. The discussion should identify any changes related to the power uprate with regard to component inspection and testing programs, and erosion/corrosion programs, and discuss the significance of these changes. If changes are insignificant, the licensee should explicitly state so.

F. The discussion should address whether the effect of the power uprate on steam generator tube cycle fatigue is consistent with NRC Bulletin 88-02, *Rapidly Propagating Fatigue Cracks in Steam Generator Tubes,* February 5, 1988.

RESPONSE TO IV. MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

IV.1.A.i Reactor Vessel

The reactor vessel stress and fatigue usage factors were evaluated at the uprated operating conditions. The evaluation assessed the effects of the revised operating parameters on the most limiting locations. The reactor vessel components were originally analyzed for a T_{cold} of 536.6°F and a T_{hot} of 623.2°F. Comparing the uprate operating conditions to the analysis of record, T_{cold} decreased by 0.6°F to 536.0°F and T_{hot} increased by 0.6°F to 623.8°F. These changes in temperature only affect the unit loading and unloading (15% to 100%) transients for the reactor vessel. All other transients remain bounded by those for current operation. The unit loading and unloading transients were evaluated for the changes in T_{cold} and T_{hot} resulting from the power uprate. The evaluation compared the current analysis of record and power uprate transient thermal stresses created by the unit loading and unloading transients on two representative reactor vessel wall thicknesses. The wall thicknesses selected pertained to the reactor vessel shell nozzle beltline region and the outlet nozzle reinforced region. There was an insignificant change in the transient thermal stresses between the analysis of record and power uprate conditions for these wall thicknesses. Since the transient pressure stresses and external mechanical load stresses did not change at the uprated conditions, the stress and fatigue results for the current analysis of record remain bounding.

CUF, including environmental effects, were calculated for the lower shell juncture, inlet nozzles and outlet nozzles per the locations identified for Newer Vintage Westinghouse reactor vessels in NUREG/CR-6260 (Reference IV-1). The CUF were calculated based on transient thermal stresses and an environmental fatigue factor. These calculated CUFs are all less than the 1.0 limit with substantial margin.

The code of record is listed in Section IV.1.D and remains unchanged. The reactor vessel components continue to meet the stress and fatigue limits specified in ASME B&PV Code, Section III, for plant operation at the uprated power conditions.

IV.1.A.i.1 Primary Shield Wall Gamma Heating

The primary shield wall is located adjacent to the reactor vessel. A revised gamma dose was calculated for the primary shield wall at 55 EFPY (EOL). This revised gamma dose was used to determine the impact on the primary shield wall evaluation previously reported in the HNP license renewal application (Reference IV-17). The revised gamma heating rates were determined by taking the values reported in the license renewal application and multiplying them by a conservative ratio. Specifically, a 4% increase in Cycles 11-17 gamma heating rate was applied to Cycles 18+ to conservatively bound the actual conditions resulting from the power uprate.

The maximum projected gamma dose at the primary shield wall inside surface at 55 EFPY is 1.33 E10 rads following the power uprate, compared to the pre-uprate value of 1.29 E10 rads. This increase in gamma dose is small. The HNP Structural Monitoring Program requires inspections of the primary shield wall accessible concrete at least every 10 years, so the change in projected gamma dose should not result in an aging effect requiring management. The power uprate conditions are bounded by the current licensing basis.

IV.1.A.ii Reactor Vessel Internals

The revised design conditions were evaluated for impact on the existing reactor vessel internals design basis analyses, as follows:

IV.1.A.ii.1 Core Bypass Flow

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. The current design core bypass flow limit is 7.1% of the total reactor vessel flow with thimble plugging devises installed and 8.6% with thimble plugging devices removed. Using the same fuel assembly hydraulic characteristics and system parameters; such as inlet temperature, reactor coolant pressure and flow; the THRIVE code was used to determine the power uprate effect on the total core bypass flow. The power uprate maximum core bypass flow value was calculated with thimble plugging devices removed and installed. The existing design core bypass flow value of 7.1% of the total reactor vessel flow with thimble plugging devises installed and 8.6% with thimble plugging devices removed remains bounding at the MUR power uprate conditions.

IV.1.A.ii.2 Rod Control Cluster Assembly Drop Time

The revised design conditions were evaluated for impact on the RCCA drop time. Only the lowest core inlet temperature was evaluated, since fluid density is greatest at the lower temperature and therefore the limiting condition for RCCA drop time. The effect of increased core power with a decreased core inlet temperature of 0.6°F has an insignificant (approximately 0.003 second increase) impact on RCCA drop time. The RCCA drop time at MUR power uprate conditions (2.41 seconds) remains below the TS limit of 2.7 seconds and is acceptable.

IV.1.A.ii.3 Hydraulic Lift Forces

The reactor internals hold-down spring provides a restraining force against the core barrel lifting due to hydraulic forces. This restraining force also ensures that an adequate friction force is maintained at the interface so sliding does not occur between the core barrel and reactor vessel. An evaluation was performed to determine the total hydraulic lift on the lower reactor internals, and assess the power uprate impact on the reactor internals hold-down spring. The evaluation determined that the impact on total hydraulic lift force is less than a 0.2% increase, and the hold-down spring would continue to provide an adequate restraining force against core barrel lifting.

IV.1.A.ii.4 Mechanical Evaluation

The reactor vessel internals are designed to withstand forces resulting from mechanical loads generated by a LOCA, seismic event, and flow-induced vibration. The revised design conditions do not affect the current design basis for LOCA, seismic or flow-induced vibration loads. The existing LOCA analysis results are applicable at power uprate conditions. The seismic response spectra are unaffected by the power uprate and remain applicable. Flow-induced vibration can be influenced by changes in vessel inlet temperature, vessel outlet temperature, and hydraulic design parameters. The design power capability parameters for the current design basis and the power uprate are essentially the same. Therefore, there is no significant impact on reactor internals performance with regard to flow-induced vibration.

IV.1.A.ii.5 Structural Evaluation

Evaluations were performed to demonstrate that the structural integrity of reactor internal components is not adversely affected by the power uprate. The reactor internal components that are subjected to heat generation effects, either directly or indirectly are the upper core plate, lower core plate, core baffle plates, former plates, core barrel, thermal shield, baffle-former bolts, and barrel-former bolts. For reactor internal components, except the lower core plate, upper core plate, and the baffle-former bolts, the stresses and cumulative fatigue factors of the previous analyses remain bounding at power uprate conditions.

IV.1.A.ii.5.a Lower Core Plate Structural Analysis

The lower core plate is subjected to heat generation effects due to its proximity to the core. Structural evaluations were performed to demonstrate that the lower core plate structural integrity was not adversely affected by the revised design conditions. The lower core plate maximum primary plus secondary stress intensity and cumulative usage factor, including the effects of increased heat generation rates, are acceptable. The lower core plate is structurally adequate for the MUR power uprate conditions.

IV.1.A.ii.5.b Baffle-Former Bolt Evaluation

The baffle-barrel region consists of a core barrel with installed baffle plates. Bolting connects former plates to the baffle plates and core barrel. The baffle-to-former bolts restrain baffle plate motion. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, LOCA and seismic loads; secondary loads consisting of preloads; and thermal loads resulting from RCS temperatures and gamma heating rates. The baffle-to-former bolt thermal loads are induced by differences in the average metal temperature between the core barrel and baffle plate. In addition to providing structural restraint, the baffles channel and direct coolant flow.

Structural qualification of the baffle-former bolts was accomplished by comparing the HNP baffle-bolt data with that from Almaraz Unit 2 (Reference IV-2), which has the same baffle-barrel and baffle-former bolt designs. A comparison of plant operating parameters indicated that the two plants are similar; the small differences would have an insignificant impact on the thermal analysis used to generate the baffle plate differential temperatures. Therefore, the existing Almaraz Unit 2 analysis is applicable to HNP, with a cumulative usage factor less than the 1.0 limit. The baffle-former bolts are structurally adequate for the uprated conditions.

IV.1.A.ii.5.c Upper Core Plate Structural Analysis

The upper core plate positions the upper end of the fuel assemblies and the lower ends of the control rod guide tubes. It also controls coolant flow exit from the fuel assemblies and serves as a boundary between the core and the exit plenum.

The maximum stress contributor in the upper core plate results from the average temperature difference between the center portion of the upper plate and the rim. The increased stress from the increased gamma heating was determined, and the results indicate that the upper core plate structural integrity is maintained at power uprate conditions. The upper core plate maximum primary plus secondary stress intensity and cumulative usage factor, including the effects of increased heat generation rates, are acceptable. The upper core plate is structurally adequate for the uprated conditions.

IV.1.A.iii Control Rod Drive Mechanism

The CRDMs use electro-magnetic coils to position the RCCA within the reactor core. The revised design conditions were reviewed for impact on the existing CRDM design basis analyses. The major inputs that can be affected by the power uprate are cold leg temperature (T_{cold}), the NSSS design transients, and the LOCA loads. The uprate T_{cold} value of 553.8°F is bounded by the T_{cold} value of 557.4°F used in the analysis of record. The NSSS design transients in the analysis of record remain applicable at uprate conditions. The LOCA loads are unchanged. Therefore, the CRDM pressure boundary remains bounded by the analysis of record at power uprate conditions.

Westinghouse Nuclear Safety Advisory Letter NSAL-07-3 (Reference IV-3) identified underestimation in the seismic loads used to qualify the CRDM head adapters at various nuclear plants, including HNP. The underestimated loads apply to the CRDM pressure housing. For the CRDM pressure housing faulted structural evaluation, the faulted bending moments are compared to allowable limits along the entire length of the CRDM pressure housing. Seismic loads were adjusted to account for the underestimated loads used in the analysis of record. The new loads remain bounded by the allowable and are therefore acceptable. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.iv Reactor Coolant Piping and Supports

The revised design conditions were evaluated to determine the impact on the existing as-built design basis reactor coolant loop piping system analysis, and the pressurizer surge line piping thermal stratification analysis for the following:

- Reactor coolant loop piping stresses and displacements
- Pressurizer surge line piping (including effects of thermal stratification and fatigue)
- Primary equipment nozzles (reactor pressure vessel inlet and outlet, SG inlet and outlet, and RCP suction and discharge)
- Primary equipment supports (reactor pressure vessel supports, SG columns and lateral bumpers, RCP columns and tie rods, and pressurizer supports)
- Reactor coolant loop branch nozzles
- Class 1 auxiliary piping systems attached to the reactor coolant loop
- Loads used in leak-before-break analysis
- Reactor coolant loop fatigue evaluation

The evaluations performed concluded that the current analyses of record remain valid for reactor coolant loop piping and supports, Class 1 auxiliary lines and branch nozzles, and surge line stratification (including fatigue). The reactor coolant piping and supports remain acceptable at MUR power uprate conditions. The codes of record are listed in Section IV.1.D and remain unchanged.

IV.1.A.iv.1 Deadweight Analysis

The existing deadweight analysis remains applicable at power uprate conditions, since the piping and equipment weight does not change.

IV.1.A.iv.2 Thermal Analysis

Reactor coolant loop temperatures for all three legs (hot leg, cross-over leg and cold leg) are either less than the current design basis reactor coolant loop temperatures, or increase by an amount that will not have a significant impact on the thermal analysis. The applicable reactor coolant loop thermal piping loads, thermal piping stresses, primary equipment thermal support loads and displacements, primary equipment thermal nozzle loads, and the current design basis thermal loads and displacements at the Class 1 auxiliary piping line connections to the reactor coolant loop remain applicable at power uprate conditions.

IV.1.A.iv.3 Seismic and LOCA/Pipe Break Analyses

There are no changes to the piping system mass or the response spectra, so the seismic analysis is unchanged. There are no changes to the LOCA hydraulic forces. Since the hydraulic forces remain unchanged, the reactor vessel motions also remain unchanged. Therefore the existing LOCA analysis remains valid at power uprate conditions. The main steam and feedwater break analyses of record remain valid for uprate conditions.

IV.1.A.iv.4 Reactor Coolant Loop Fatigue Analysis, Leak-Before-Break Loads, and Primary Equipment Nozzle Loads

The existing primary side reactor coolant loop NSSS design transients do not change. There are no changes to the reactor coolant loop deadweight, thermal, and seismic analyses, so the existing reactor coolant loop fatigue analysis and leak-before-break loads do not change. Since there are no changes to the reactor coolant loop deadweight, thermal, seismic, and LOCA/pipe break loading conditions, the primary equipment supports and primary equipment nozzle loads do not change and the analysis of record remains valid at uprate conditions.

IV.1.A.iv.5 Reactor Coolant Loop Primary Equipment Supports

The reactor coolant loop deadweight, thermal, seismic, and LOCA/pipe break analyses remain valid at power uprate conditions. Since the support load inputs are unchanged and remain valid, the primary equipment supports are acceptable and the primary equipment support stress values remain applicable.

IV.1.A.iv.6 Class 1 Auxiliary Line Analysis, including Branch Nozzle Forces and Displacements

There are no changes to the reactor coolant loop deadweight, thermal, seismic, LOCA/pipe break, and fatigue analysis inputs at uprate conditions, so there is no change to the Class 1 auxiliary line piping and auxiliary branch nozzle analyses and they remain valid. There is no change to the auxiliary line branch nozzle forces and displacements.

IV.1.A.iv.7 Reactor Vessel Outlet Nozzle (MSIP)

A Mechanical Stress Improvement Process (MSIP) was implemented on reactor pressure vessel outlet nozzles during Refueling Outage 16 (Fall 2010). The MSIP equipment applies a narrow permanent radial deformation adjacent to a piping weld that redistributes the as-welded residual stresses in the weld, with a zone of compressive residual stresses at the inner region of the weld joint. This deformation results in pipe (hot leg) expansion in the axial direction, similar to thermal expansion. The MSIP axial expansion is evaluated as a residual stress according to the cold-spring criteria in ASME B&PV, Section III, NB-3672.8. The pipe stress, support loads, and nozzle loads were evaluated for the MSIP axial expansion and shown to be acceptable. Fatigue at the reactor pressure vessel nozzle-to-pipe weld was evaluated. The fatigue evaluation is applicable to power uprate conditions. There is no impact to the auxiliary branch nozzle displacements due to the MSIP.

IV.1.A.iv.8 Pressurizer Surge Line Analysis

Surge line stratification is not significantly affected by the power uprate, since the hot leg temperature increases slightly (0.6°F), which reduces surge line stratification. There is no change to the NSSS design transients. Therefore, the current analysis of record for surge line stratification remains valid at power uprate conditions. There are no changes to the reactor coolant loop deadweight, thermal, seismic, LOCA, and fatigue analysis inputs at uprate conditions, so the existing surge line piping analyses do not change.

IV.1.A.v Balance-of-Plant Piping (NSSS Interface Systems, Safety-Related Cooling Water Systems and Containment Systems) and Supports

BOP system operation at uprate conditions may result in increased piping stress levels, piping support loads, nozzle loads, etc. due to higher system operating temperatures, pressures and flow rates. BOP piping and support systems were evaluated at power uprate operating conditions to ensure that they will continue to perform their intended functions. The evaluations also demonstrate design basis compliance with ASME B&PV, Section III, Division 1, 1971 Edition including all addenda through the Summer 1973 and ANSI B31.1 Power Piping Code 1973

Edition, including the Summer 1973 Addendum. The following piping and support systems were evaluated:

- Auxiliary Feedwater
- Auxiliary Steam
- Steam Generator Blowdown
- Component Cooling Water
- Condensate
- Extraction Steam
- Feedwater
- Heater Vents and Drains
- Main Steam
- Service Water
- Spent Fuel Pool Cooling & Cleanup
- Chemical & Volume Control
- Residual Heat Removal
- Safety Injection
- Containment Spray

Thermal, pressure, and flow rate change factors were determined, as required, to compare and evaluate the changes in operating conditions. These change factors were based on the following ratios:

- The thermal change factor was based on the ratio of power uprate to pre-uprate operating temperature $(T_{UPRATE} - 70°F) / (T_{ANALYZED} - 70°F)$.
- The pressure change factor was determined by the ratio of $(P_{UPRATE} / P_{ANALYZED})$.
- The mass flow rate (MFR) change factor was determined (for those systems currently analyzed for fluid transient events) by the ratio of $(MFR_{UPRATE} / MFR_{ANALYZED})$.

These thermal, pressure and flow rate change factors were used in determining piping systems acceptability for power uprate conditions. When a change factor is less than or equal to 1.0 (the existing pre-uprate condition envelops or equals the power uprate condition), the piping system is considered acceptable for power uprate conditions. When the change factor is greater than 1.0, an evaluation is performed for the specific increase in temperature, pressure and/or flow rate, to document pipe stress design basis compliance.

The following change factors were greater than 1.0: portions of condensate system (1.02 to 1.18), portions of extraction steam system (1.01), portions of feedwater system (1.02), portions of heater vents and drains (1.00 to 1.04). These were thermal change factors based on increased operating temperatures. The resulting stress levels were evaluated for these systems and demonstrated to be within acceptable stress limits in all cases.
The evaluation results demonstrate that the BOP piping and pipe support systems reviewed continue to satisfy the design basis requirements when considering the temperature, pressure and flow rate effects resulting from the power uprate. BOP piping and support systems modifications are not required. The BOP piping and pipe support systems reviewed remain acceptable at MUR power uprate conditions.

IV.1.A.vi Steam Generator

The original Model D4 SGs were replaced in 2001with Model Delta-75 SGs. This was a complete SG replacement with no parts retained from the original SGs. The codes of record are listed in Section IV.1.D and remain unchanged.

IV.1.A.vi.1 Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluation focused on changes to secondary side operating characteristics at power uprate conditions. SG secondary side performance characteristics such as steam pressure and flow, circulation ratio, bundle mix flow, heat flux, secondary side pressure drop, moisture carryover, hydrodynamic stability, secondary side mass and other parameters are affected by increases in power level. Secondary side performance characteristics (except DNB) were calculated using the SG performance code GENF. GENF code analyses were performed for the design parameter cases. A separate analysis was performed using the 3-D code ATHOS (DNB) to determine the detailed flow parameters throughout the tube bundle. The thermal-hydraulic evaluation concluded that the SG thermal-hydraulic operating characteristics remain acceptable for the MUR power uprate.

IV.1.A.vi.2 Structural Integrity

The structural evaluation focused on the critical SG components as determined by the design basis analyses stresses and fatigue usages.

To determine the effects of the power uprate, scale factors were conservatively developed to extend the existing analysis. Scale factors are calculated based on the ratio of the uprate conditions to the previous conditions, and are then used to update stress range and fatigue usage. Only critical SG components or areas are examined to evaluate the power uprate effect on the component stresses and fatigue usage. Less critical SG areas are enveloped by the evaluation. The primary side limiting area is located at the junction of the channel head to the tubesheet. The stresses for the primary side analysis are proportional to the differential pressure between the primary and secondary side. The minor shell taps were selected to envelop the secondary side items because they have the most limiting value for fatigue usage. The secondary side components are exposed to the secondary side pressure. Therefore, the stresses are proportional to the steam pressure. As a conservative measure, the change in steam pressure has been used to develop scale factors for the stress ranges. The fatigue usage of all primary side and secondary side components remains below 1.0 over the plant design life. All the critical SG components are acceptable for operation at uprated conditions.

An analysis was performed to determine if the ASME B&PV Code limits on design primary-to-secondary differential pressure are exceeded for any applicable transient at power uprate conditions. The analysis determined that the maximum primary-to-secondary side differential pressures during normal operating transients, assuming 10% SG tube plugging, are 1456 psi and 1494 psi for high T_{avg} and low T_{avg} temperatures respectively. The maximum primary-to-secondary side differential pressures during upset condition transients, assuming 10% SG tube plugging, are 1470 psi and 1742 psi for high T_{avg} and low T_{avg} temperatures respectively. These values are below the applicable design pressure limits of 1600 psi for normal operating transients and 1760 psi for upset condition transients. Therefore, the ASME B&PV Code limits on primary-to-secondary differential pressure are satisfied at MUR power uprate conditions.

IV.1.A.vi.3 Tube Bundle Integrity, Flow Induced Vibration and Wear

IV.1.A.vi.3.a Tube Integrity

The Model Delta-75 replacement SGs contain thermally treated Alloy 690TT tubes and Type 405 stainless steel tube support plates with broached trefoil holes. The trefoil tube hole configuration results in reduced potential for contaminant concentration at tube support plate intersections by reducing the crevice area. The first eight tubes rows were heat treated after bending to relieve stresses. Thermally treated Alloy 690TT is highly resistant to stress corrosion cracking. The replacement SGs have exhibited no indications of corrosion related tube degradation after five cycles of operation. Actual tube plugging levels are essentially 0% since the replacement SGs were installed. Seven SG tubes have been plugged (SGA-3, SGB-1, and SGC-3), two were plugged pre-service and the remaining five were plugged due to loose parts wear. No active systematic corrosion mechanisms have been identified. During SG condition monitoring and operational assessment evaluations prepared after each SG inspection, only foreign objects/loose parts wear was identified as an existing SG tube degradation mechanism. Potential mechanisms such as anti-vibration bar wear and outside diameter stress corrosion cracking were absent, but are included in the inspection planning. To date, the Model Delta-75 SGs have had no incidence of primary water stress corrosion cracking.

After the power uprate, potential tube degradation mechanisms resulting from hypothetical localized chemistry changes at the tube surfaces are the various modes of outside diameter stress corrosion cracking. Based on laboratory and operating experience, and current operating and maintenance practices, the power uprate will not produce excessive degradation due to outside diameter stress corrosion cracking. On the basis of temperature increase alone, the mechanical wear processes are unlikely to be significantly changed. No effect on the incidence of primary water stress corrosion cracking is expected, since Alloy 690TT is highly resistant to primary water stress corrosion cracking.

IV.1.A.vi.3.b Flow Induced Vibration and Wear

SG tube wear (i.e., fretting) was evaluated based on current design basis analysis and consideration of SG secondary side thermal-hydraulic changes resulting from the power uprate. SG tube wear due to fluid-elastic effects in the U-bend region and turbulence induced displacement effects in the straight leg tube region were considered.

The analysis results indicate an increase in fluid-elastic stability ratio by as much as 3.4%, with an increase in vibration amplitude due to turbulence by as much as 6.9%. The tube stability ratio increase of 3.4% results in a stability ratio of 0.42, which is less than the 1.0 allowable. Increasing the baseline vibration amplitude by 6.9% results in 17.4 mils of amplitude, which is less than one-half the distance separating the tubes (146 mils). Both conditions remain acceptable following the power uprate. The maximum pre-uprate predicted SG tube wear is 4.0 mils over the SG 40-year design life. The power uprate increases the tube wear by 6.9% over the calculated original design power level. This results in a maximum post-uprate predicted SG tube wear of 4.9 mils over the projected 60-year plant life. This value is below the tube plugging limit of 16 mils (40% wear depth). The amount of tube wear will not significantly affect tube integrity and is acceptable.

Other items reviewed were tube stress and fatigue. Tube stress resulting from flow induced vibration concerns after the power uprate is less than 2.0 ksi. This results in a corresponding fatigue usage of 0.024, which is less than the 1.0 allowable. Therefore, tube stresses are acceptable at MUR power uprate conditions, the flow induced vibration loading fatigue usage factor is negligible, and fatigue degradation from flow induced vibration is not anticipated

IV.1.A.vi.4 Steam Drum Evaluation

The dominant factors affecting FAC in the SG steam drum region are the material composition and fluid velocity. The steam drum components of concern are the feedwater ring, spray nozzles, and the primary moisture separator assemblies; because these are the primary locations where other plants have experienced FAC. Secondary separators were also addressed even though degradation of these components has not been observed at other plants. Operation at uprated plant conditions will increase feedwater flow rates in the SGs, with the possibility of initiating, increasing, and/or accelerating FAC in any susceptible component within the steam drum region.

IV.1.A.vi.4.a Feedwater Ring, Spray Nozzles and Primary Separators

The FAC rate after the power uprate is predicted to be no more than 6% above the current conditions. The predicted 0.89 mil/year degradation at uprate conditions would result in a material loss of approximately 0.05 inches over 60 years of plant operation. This is a small fraction of the primary separator vanes and risers thickness. The feedwater velocity increase due to the uprate is less than 2.0%, which has no impact on the potential for feedwater nozzle erosion. The SG feedwater rings were inspected during Refueling Outage 13 (Spring 2006), and the only degradation detected was some slight wear caused by foreign objects trying to pass through the spray nozzles. SGB steam drum components were also visually inspected during Refueling Outage 13. The overall steam drum condition was excellent, with no observed wear or degradation.

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IV.1.A.vi.4.b Secondary Separators

The secondary separator vanes are fabricated from A366 carbon steel and would be susceptible to FAC at sufficient fluid velocities. The velocities at uprate conditions are well below the threshold for significant FAC. Therefore, secondary separator degradation is not expected.

The SG construction materials, which include chromium, were selected for their resistance to FAC. The fluid velocities in the primary separators have not increased sufficiently to cause a significant increase in FAC rate at power uprate conditions. Secondary separator degradation is not expected due to the low fluid velocities. SG inspections conducted after five years of operation identified no evidence of FAC. Therefore, the effect of the MUR power uprate on the FAC rate, and the likelihood of significant future FAC are minimal.

IV.1.A.vi.5 Mechanical Repair Hardware

Mechanical repair hardware refers to components such as plugs, sleeves, and stabilizers that are installed in the SGs to address tube degradation.

IV.1.A.vi.5.a Shop Weld Plugs

The critical parameter affecting the plug design is primary-to-secondary differential pressure. The maximum differential pressure of 3107 psig produced a stress-to-allowable ration of 0.60 for the primary side hydrostatic pressure test. This pressure bounds all normal, upset, emergency, and faulted conditions for the power uprate. The six fatigue exemption conditions in ASME B&PV, Section III, NB-3222.4 (d) are satisfied for the shop installed weld plugs. Since all six fatigue exemption conditions are satisfied, an explicit fatigue analysis is not required. The shop installed weld plugs continue to satisfy all ASME B&PV Code structural limits and remain qualified for use in the Model Delta-75 SGA and SGB. There are no shop installed weld plugs in SGC.

IV.1.A.vi.5.b Ribbed Mechanical Plugs

Plug retention margin is available for the most limiting retention conditions. In all cases, the ratio of calculated stress intensity to the ASME B&PV Code allowable stress intensities is less than 1.0. The six fatigue exemption conditions in ASME B&PV Code, Section III, NB-3222.4 (d) are satisfied for the ribbed mechanical plugs. Since all six fatigue exemption conditions are satisfied, an explicit fatigue analysis is not required. The analyses performed show that both the long and short 11/16-inch ribbed mechanical plug designs satisfy applicable stress and retention acceptance criteria at power uprate conditions and remain qualified for use in the HNP SGs.

IV.1.A.vi.5.c Roll Expanded Mechanical Plugs

The roll expanded mechanical tube plug was initially qualified by performing a test program demonstrating that the tube plug satisfied the intent of ASME B&PV Code, Section III with regard to primary stress (pressure), secondary stress (thermal), and fatigue (tubesheet flexure). A detailed review of the roll expanded mechanical tube plug qualification data, and operating and design data was performed. The limiting criterion for qualification at power uprate conditions is primary-to-secondary differential pressure. The qualification test reports are bounding for primary-to-secondary differential pressure. Since there are no changes in tube plug design or transient conditions identified in the design specifications, the thermal stress and tubesheet flexure evaluation is unaffected and bounded for the design values. The test program used to qualify the roll expanded mechanical tube plugs installed in the SGs remains valid at MUR power uprate conditions.

IV.1.A.vi.5.d Cable Stabilizer

These components were initially designed for SG tubes that had circumferential cracks near the expansion-transition zone near the top of the tubesheet. However, they may be used in the straight-leg tube region at any location in the tube bundle where tube integrity could be compromised. The only stabilizer parameters affected by the power uprate are stability ratio and tube displacements. Both of these parameters are within specified acceptance criteria at power uprate conditions. Therefore, it is acceptable to install straight-leg cable stabilizers at any location in the SG tube bundle straight legs.

IV.1.A.vi.6 Loose Parts

There are no foreign objects present in SGB or SGC. Foreign object search and retrieval operations during previous refueling outages determined that only minor objects remain in SGA.

The previous loose parts evaluation was reviewed to determine the power uprate effects on the projected wear times. The wear time analysis was performed assuming 20% initial tube wear. The SG secondary side conditions will change as a result of the power uprate operating conditions, however, these changes do not affect the previous evaluation conclusions.

A foreign object that is predicted to have a low wear value (i.e., below the structural limit of 57.5% wear) over two cycles of operation can be classified as a minor object. The analysis determined by the flow conditions ratio method, that any increase in wear environment severity is minor due to the power uprate. The SG can withstand three cycles of operation with the documented loose parts that remain on the secondary side, without reaching the limiting tube wall wear value. Therefore, SG operation is acceptable for three cycles between inspections when operating at MUR power uprate conditions.

IV.1.A.vi.7 Regulatory Guide 1.121 Analysis

RG 1.121 describes an acceptable method for establishing the limiting safe tube degradation beyond which tubes found defective by inservice inspection must be repaired or removed from service. The acceptable degradation level is called the repair limit.

The RG 1.121 evaluation defines the structural limit for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. SG tubing structural limits were determined by analysis (Reference IV-4), for an assumed uniform thinning degradation mode in both the axial and circumferential directions. The existing analysis is applicable to power uprate conditions. Conservative design criteria have been established for maintaining tube structural integrity under the postulated design basis accident condition loadings, in accordance with ASME B&PV Code, Section III, 1971 minimum strength properties.

The allowable tube repair limit is determined by adjusting the structural limit per RG 1.121 to take into account uncertainties in eddy current measurement, and an operational allowance for continued tube degradation until the next scheduled inspection. Analyses have been performed to establish the structural limit for the tube straight-leg (free span) region for degradation over an unlimited axial extent, and for degradation over a limited axial extent at the tube support plate and anti-vibration intersections. The current structural limit is not changed by the power uprate. The existing tube repair limit is unaffected by the MUR power uprate and remains valid.

IV.1.A.vii Reactor Coolant Pumps and Reactor Coolant Pump Motors

Revised RCS conditions were reviewed for impact on the existing RCP design basis analysis. The NSSS design parameters considered in the RCP evaluation are SG outlet temperature and RCS pressure. No changes in RCS design or operating pressure were made as part of the power uprate. The maximum SG outlet temperature for any NSSS design parameters case is 553.5°F.

This temperature is lower than the previously evaluated SG outlet temperature of 557.1°F. Due to lower allowable design stress limits, higher temperatures are more limiting for RCP structural design qualification and the NSSS parameter change for the power uprate is therefore conservative. The uprated conditions remain bounded by the original design conditions or previously evaluated conditions. The NSSS design transients previously used in the RCP component fatigue analyses remain applicable at power uprate conditions. There are no changes to the temperature variations, pressure variations, or number of occurrences for any of these transients. RCP pressure boundary components continue to meet stress and fatigue requirements at power uprate conditions.

The RCP motor loading was evaluated in four areas: continuous hot loop, continuous cold loop, starting and thrust bearing loads. The worst hot loop condition occurred at the highest SGTP, resulting in a maximum pump brake hp of 6952, which is bounded by the RCP motor nameplate rating of 7000 brake hp. There is no impact on cold loop conditions or operating parameters that affect the motor starting evaluation. The thrust bearing loading remains acceptable for hot and cold conditions. The RCP motors are therefore acceptable at MUR power uprate conditions.

IV.1.A.viii Pressurizer

The pressurizer meets the stress/fatigue analysis requirements for plant operation at the MUR power uprate conditions. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.viii.1 Structural Evaluation

The revised operating conditions were reviewed for impact on the existing pressurizer design basis analyses. The limiting pressurizer operating conditions occur when the RCS pressure is high and the RCS T_{hot} and T_{cold} are low. No changes were made in RCS design or operating pressure as part of the power uprate. The minimum T_{hot} and T_{cold} values from the design parameter cases were used in the pressurizer evaluation. The existing analysis of record justified pressurizer operation with T_{hot} temperatures within a range of 605.9°F to 623.2°F and T_{cold} temperatures within a range of 536.6°F to 555.5°F. At the normal operating pressure of 2250 psia, the revised T_{hot} temperature for normal operation is bounded by the analysis. The T_{cold} is 0.6°F lower than the analysis of record. However, the magnitude of the difference is within the analytical tolerances for the design transient analyses.

The existing NSSS design transients were evaluated for the uprated conditions. The evaluation concluded that the existing NSSS design transients (transient description, total number of occurrences for 40 years of plant operation and transient curves) remain valid at power uprate conditions. The existing NSSS design transients were also evaluated for continued applicability to 60 years of plant operation. The total number of occurrences for 40 years of plant operation remains valid and conservative for 60 years of operation. Since the existing number of transient cycles, NSSS design transient descriptions, and plots remain valid at uprate conditions, the existing design transients continue to remain valid at power uprate conditions for 60 years of plant operation.

The pressurizer spray evaluation considered an increase in maximum spray flow rate up to 1400 gpm. Only the spray nozzle and other pressurizer components that may be affected by the increased spray flow rate required evaluation for the power uprate. The fatigue usage for the critical section in the spray nozzle was recalculated to account for the higher spray flow rate and remains below the ASME B&PV Code limit of 1.0. The only transients that affect the upper shell and trunnion buildup are plant heatup and cooldown. The current fatigue usages remain valid for the upper shell and trunnion buildup. The highest fatigue usage in the surge nozzle was at the safe end, which has been covered by a structural weld overlay (refer to discussion below). The highest fatigue usage outside the overlay region was at the nozzle knuckle. The fatigue usage at this location was revised and remains below the ASME B&PV Code limit of 1.0.

The pressurizer lower head and surge nozzle were previously evaluated for insurge/outsurge transient effects related to both design transients and operational transients that were not considered in the original design. The revised design parameters were evaluated for their effect on the previous evaluation conclusions. The revised design parameters have an insignificant impact on the previous fatigue results and they remain valid. The pressurizer lower head and surge nozzle meet the stress and fatigue analysis requirements when subjected to loadings, including insurge/outsurge transients, at power uprate conditions. The analyses of record for insurge/outsurge and related environmental fatigue remain unchanged.

The spray nozzle and surge nozzle were the only fatigue usage changes resulting from the uprate. The spray nozzle change was due to the increase in maximum spray flow rate. The surge nozzle change was because the section under the weld overlay was replaced with the section through the nozzle knuckle. The ASME B&PV Code stress limits remain satisfied for all pressurizer components at power uprate conditions.

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IV.1.A.viii.2 Structural Weld Overlay Evaluation

Preemptive full structural weld overlays have been applied to the safe ends of the pressurizer nozzles (surge, safety/relief valve, and spray line) to eliminate dependence upon the Alloy 82/182 welds as pressure boundary welds, and to mitigate any potential primary water stress corrosion cracking in these welds.

The power uprate effect on these structural weld overlays was evaluated. The power uprate does not change any of the current design transient curves for the pressurizer surge, safety/relief valve, or spray line nozzles. The nozzle loads (forces/moments) are equivalent to or bounded by those analyzed during the weld overlay analyses. The pressurizer temperatures and pressures are equivalent to or bounded by those analyzed during the weld overlay analyses. The insurge/outsurge transients for the surge nozzle remain unchanged at uprate conditions. The RCS cold leg temperature decreases by 0.6°F. This temperature decrease is within the analytical tolerances for the design transient analysis and does not adversely impact the pressurizer or attached components; no evaluation is required for this condition.

The surge nozzle and safety/relief valve nozzle weld overlay analyses are not adversely affected by the power uprate conditions. These locations continue to meet the ASME B&PV Code, Section III allowable and fatigue usage requirements.

The original spray nozzle analyses used a design flow rate lower than the MUR maximum spray flow rate of 1400 gpm. The impact of this increase in maximum flow rate through the spray nozzle was evaluated for the spray nozzle weld overlay configuration. There is sufficient margin in the analyses to accommodate the spray flow increase. The spray nozzle weld overlay analyses continue to meet the ASME B&PV Code, Section III allowable and fatigue usage requirements.

Therefore, the pressurizer surge, safety/relief valve, and spray line nozzles are qualified for MUR power uprate conditions.

IV.1.A.ix Safety-Related Valves

The revised design conditions were reviewed for impact on the existing safety-related valves design basis analyses. No changes in RCS design or operating pressure were made as part of the power uprate. The evaluations concluded that the temperature changes due to the power uprate are bounded by those used in the existing analyses. Safety-related valves were reviewed within the applicable system (Section VI) and program (Section VII.6.E) evaluations. None of the safety-related valves required a change to their design or operation as a result of the MUR power uprate.

IV.1.B.i Stresses

The revised design conditions for the NSSS components and BOP piping (NSSS interface systems, safety-related cooling water systems and containment systems) were reviewed for impact on the existing design basis analyses. Structural evaluations (stress and cumulative usage factors) are discussed in Sections IV.1.A.i (reactor vessel), IV.1.A.ii (reactor vessel internals), IV.1.A.iii (control rod drive mechanism), IV.1.A.iv (reactor coolant piping and supports), IV.1.A.vi (steam generator), IV.1.A.vii (reactor coolant pumps and motors), IV.1.A.viii (pressurizer), and IV.1.A.ix (safety-related valves). No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes are within design limits. The evaluations reviewed maximum stress intensities/stress ranges, with comparison to stress allowable, cumulative usage factors (for Class 1), and other special stress limits.

IV.1.B.ii Cumulative Usage Factors

The revised design conditions for the NSSS components, piping, and interface systems were reviewed for impact on the existing design basis analyses. Structural evaluations (stress and CUF) are discussed in Sections IV.1.A.i (reactor vessel), IV.1.A.ii (reactor vessel internals), IV.1.A.iii (control rod drive mechanism), IV.1.A.iv (reactor coolant piping and supports), IV.1.A.vi (steam generator), IV.1.A.vii (reactor coolant pumps and motors), IV.1.A.viii (pressurizer), and IV.1.A.ix (safety-related valves). The Class 1 fatigue analyses remain valid for the 60-year plant life. The CUFs remain below the ASME B&PV Code allowable value of 1.0. The environmentally assisted fatigue analyses evaluated the impact of the reactor coolant environment on the CUF. The current environmentally assisted fatigue evaluations were based on design specification transients, plus plant monitoring data, that did not include the power uprate changes. The power uprate was evaluated for impact on the environmental fatigue results in the analysis of record (Reference IV-18), which extended the original 40-year design life to 60 years for license renewal. A CUF considering environmentally assisted fatigue was calculated for components that were selected based on the lead indicator locations identified in NUREG/CR-6260 (Reference IV-19). The following components were evaluated:

- Reactor vessel head to shell juncture
- Reactor vessel inlet and outlet nozzles
- Pressurizer surge line nozzles
- Charging nozzles
- Safety injection nozzles
- RHR Class 1 piping location

The MUR power uprate will have a negligible impact on the environmentally assisted fatigue evaluations contained in the analysis of record. The existing environmentally assisted fatigue evaluations are valid at power uprate conditions. CUF values remain below the ASME B&PV Code allowable value of 1.0.

IV.1.B.iii Flow Induced Vibration

Reactor vessel internals flow induced vibration is discussed in Section IV.1.A.ii.4. SG flow induced vibration is discussed in Section IV.1.A.vi.3.b.

IV.1.B.iv Temperature Effects

IV.1.B.iv.1 Changes in Temperature (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the power uprate. The operating temperature changes are shown in Enclosure 1, Table 4. Specific calculation outputs include T_{hot} and T_{cold}. The current T_{avg} window has been maintained at 572.0°F to 588.8°F. There is an approximate 1.2°F increase in temperature across the core (T_{hot} increases approximately 0.6°F and T_{cold} decreases approximately 0.6°F) from current operating conditions due to the power uprate. There is no change to the RCS average temperature limit in TS 3.2.5.

Changes in main steam and feedwater system flow rates are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.iv.2 Evaluation of Potential for Thermal Stratification

NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems,* addresses thermal stresses in piping attached to the RCS that cannot be isolated. HNP previously identified the following piping lines as susceptible to thermal stratification: high head safety injection hot and cold legs, normal charging, and alternate changing. HNP chose option 3 to comply with NRC Bulletin 88-08, which required temperature instrumentation to monitor pipe temperature gradients due to containment isolation valve leakage. Outside surface mounted thermocouples were installed on the top and bottom of the pipe downstream of the check valve nearest the reactor coolant loop. The thermocouples are positioned where thermal cycling would most likely occur.

The power uprate temperature changes (T_{hot} + 0.6°F and T_{cold} - 0.6°F), when compared to current operation and evaluated using EPRI Material Reliability Program, MRP-146 (Reference IV-5) and MRP-146S (Reference IV-6), will not cause changes in the potential for cyclical thermal stratification that would require any different management approach to this issue from the existing HNP Thermal Stratification Monitoring Program. In addition, the RCS flow rates are essentially the same as the power uprate values. Thus, the effects of swirl penetration will not change due to the MUR power uprate.

HNP conducted a one-time inspection of the ASME B&PV Code Class 1 small bore piping connected to the RCS that cannot be isolated, and is susceptible to thermal stratification and intergranular stress corrosion cracking (IGSCC), as part of the License Renewal Aging Management Plan. The power uprate temperature changes, when compared to current operation, will not cause changes in the potential for cyclical thermal stratification or IGSCC that would require any different management approach to this issue from the existing Aging Management Plan.

NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification,* addresses surge line thermal stratification. Surge line thermal stratification is driven by the temperature difference between the RCS hot leg and the pressurizer. The power uprate impacts on surge line stratification, including the fatigue analysis performed for license renewal and extended plant operation (i.e., 60 years), were evaluated. The current hot leg operating temperature will increase 0.6°F from the power uprate. A higher hot leg temperature lowers the temperature differential between the hot leg and pressurizer, which reduces the stratification effects. There are no significant changes to the surge line operating conditions and therefore no significant changes to the pressurizer stratification loading. The existing design basis analyses for surge line stratification remain valid at MUR power uprate conditions.

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IV.1.B.v Changes in Pressure (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the power uprate. The nominal operating pressure is 2250 psig as shown in Enclosure 1 Table 4 and remains unchanged for the power uprate. There is no change to the RCS pressure limit in TS 3.2.5.

Changes in main steam and feedwater system pressures are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.vi Changes in Flow Rates (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the power uprate. The mechanical design RCS flow is shown in Enclosure 1 Table 4 and remains unchanged for the power uprate. There is no change to the RCS flow limit in TS 3.2.5.

Changes in main steam and feedwater system flow rates are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.vii High Energy Line Break

IV.1.B.vii.1 High Energy Line Break Locations

A review was performed to determine the power uprate impact on HELB systems. Revised operating temperatures, pressures, and flow rates were compared to the analyzed conditions. Input parameters remain bounding at the power uprate conditions. The MUR power uprate does not result in any new or revised pipe break locations, and the existing design basis analysis for pipe break, jet impingement and pipe whip remains valid.

IV.1.B.vii.2 Leak-Before-Break Evaluation

The existing leak-before-break analyses justified eliminating large primary loop pipe rupture from the structural design basis (References IV-7). The applicable pipe loadings, normal operating pressure, and temperature parameters at power uprate conditions were used in the evaluation. The leak-before-break acceptance criteria are based on SRP Section 3.6.3. The acceptance criteria are satisfied for primary loop piping at power uprate conditions. The recommended margins are satisfied, and the existing analyses conclusions remain valid.

Therefore, the dynamic effects of RCS primary loop piping breaks are not considered in the structural design basis at MUR power uprate conditions.

IV.1.B.viii LOCA Forces Including Jet Impingement and Thrust

A LOCA forces analysis generates the hydraulic forcing functions and hydraulic loads that occur on RCS components due to a postulated LOCA. No changes in RCS design or operating pressure were made as part of the power uprate. LOCA hydraulic forces increase with lower temperatures, so they are predominantly influenced by the T_{cold} modeled in the analyses. The LOCA hydraulic forces were evaluated at a minimum T_{cold} of 530°F. Other key inputs for the LOCA hydraulic forces analyses include the RCS geometry and hydraulic losses, the core barrel structural beam model, and the SG vertical divider plate structural model. None of these other key inputs is impacted by the power uprate. The LOCA hydraulic forces evaluated remain valid at MUR power uprate conditions.

IV.1.B.ix Seismic Qualification

Safety-related structures, systems and components are designed for both seismic and dynamic events as described in FSAR Sections 3.5 through 3.10. The power uprate impact on mechanical and electrical equipment seismic qualification, and the dynamic effects associated with pipe whip, jet impingement, and missile forces was evaluated. The mechanical and electrical equipment reviewed included equipment associated with systems essential to emergency reactor shutdown, containment isolation, reactor core cooling, reactor heat removal and preventing a significant release of radioactive material to the environment.

There are no changes to the seismic inputs (amplified response spectra) or loads. Seismic design is not impacted, because seismic inputs and requirements remain unchanged. Therefore, the seismic qualification of essential equipment and supports is unaffected.

The power uprate will not result in any changes to the existing missile sources, or add any components that could become a missile source. There is no impact on the existing missile barrier evaluations. No new or revised pipe break locations result from the power uprate, and there is no affect on the protection features currently in-place for protecting essential equipment from the dynamic effects of pipe whip and jet impingement. As a result, the dynamic qualification of safety-related equipment is not impacted.

The seismic qualification of mechanical and electrical equipment and the dynamic effects associated with pipe whip, jet impingement, and missile forces are not affected by the MUR power uprate conditions.

IV.1.C.i Pressurized Thermal Shock

The PTS evaluation provides a means for assessing the susceptibility of reactor vessel beltline materials to failure during a PTS event, to ensure that adequate fracture toughness exists during reactor operation. 10 CFR 50.61 provides the requirements, methods of evaluation, and safety criteria for PTS assessments.

PTS calculations were performed using the latest procedures specified in 10 CFR 50.61. New fluence projections accounting for the power uprate were developed to 36 EFPY and 55 EFPY. Calculated bounding RT_{PTS} values at 55 EFPY were determined for reactor vessel beltline and extended beltline locations (i.e., locations where fluence > 1.0E17 n/cm^2, E > 1.0 MeV). The limiting RT_{PTS} value of 209.7°F applies to the intermediate shell plate B4197-2. This limiting material is unchanged from the analysis of record.

The PTS calculations performed at 55 EFPY result in RT_{PTS} values that remain below the 10 CFR 50.61 screening criteria (270°F for plates, forgings, and axial welds; and 300°F for circumferential welds) using the projected uprate fluence. The RT_{PTS} values at MUR power uprate conditions are acceptable to 55 EFPY.

IV.1.C.ii Fluence Evaluation

Fluence calculations were based on an NRC approved methodology (Reference IV-14). This methodology follows the guidance and meets the requirements of RG 1.190 (Reference IV-15). The calculational-based fluence analysis methodology can be used to accurately predict the fast neutron fluence in the reactor vessel using surveillance capsule dosimetry, or cavity dosimetry, or both to verify the fluence predictions. The NRC approved methodology used for fluence evaluations has demonstrated that fluence analysis accuracy would be unbiased, and have a precision within the NRC suggested limit of 20%.

The NRC staff requires licensees referencing Topical Report BAW-2241P, Revision 1 in licensing applications to document how the three limitations specified in the NRC SER are met (Reference IV-16). HNP responded to these three limitations as follows:

- No features are included that differ from the evaluations that form the basis of the fluence database. Therefore, the uncertainties quoted in BAW-2241P, Revision 1 are consistent with the analysis performed.
- The cross sections utilized in the calculations are consistent with the evaluation in BAW-2241P, Revision 1.
- No modifications have been made to the methods or topical. The NRC has approved BAW-2241P, Revision 2 to apply the methods to BWRs. This revision does not impact the HNP analysis.

Revised reactor vessel fluence projections for the power uprate were calculated at 36 EFPY and 55 EFPY, and used to determine the impact on existing reactor vessel integrity evaluations. The power uprate fluence rates (fluxes) were determined by taking the values previously reported in the HNP license renewal application (analysis of record, Reference IV-17) and multiplying them by a conservative ratio. Specifically, an assumed 2% increase in Cycles 11-17 fluence rate was applied to Cycle 18+ to conservatively bound the actual conditions resulting from the power uprate (the MUR power uprate implementation is proposed for the beginning of Cycle 18). Using the conservative fluence rate assumption, the projected 55 EFPY fluence resulting from the power uprate is higher than that projected in the analysis of record. However, the increase is minor. A comparison of analysis of record peak reactor fluence and power uprate peak reactor fluence is provided below in Table IV-1.

Table IV-1
Peak Reactor Vessel Fluence

Location	Analysis of Record	MUR Power Uprate Conditions
Intermediate Shell Plate B4197-2 (limiting for RT_{PTS} and pressure-temperature limits) at base metal	55 EFPY fluence 6.80E19 n/cm^2	55 EFPY fluence 6.88E19 n/cm^2

Fluence values are used to evaluate the EOL reference temperature for PTS (RT_{PTS}), ART for developing heatup and cooldown curves and LTOP system setpoints, upper shelf energy, and transition temperature shift (ΔRT_{NDT}) for determining surveillance capsule withdrawal schedules.

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IV.1.C.iii Heatup and Cooldown Pressure/Temperature Limit Curves

ASME B&PV Code, Section XI; 10 CFR 50, Appendix G; and ASME B&PV Code Case N-640 provide the method for determining the pressure-temperature limits. These pressure-temperature limits are established to prevent non-ductile reactor vessel failure during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The controlling reactor coolant pressure boundary pressure-temperature limits at low temperatures occur in the reactor vessel beltline region. The reactor vessel materials have been tested to determine their initial RT_{NDT}. The ART is the value of initial RT_{NDT} + ΔRT_{NDT} + margins for uncertainties at a specific location. Reactor operation and resultant fast neutron (E > 1.0 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an ART based upon the fluence, copper content and nickel content of the material can be predicted and the value of ΔRT_{NDT} computed by RG 1.99, Revision 2 (Reference IV-12). The ART of the limiting beltline material is used to correct the beltline pressure-temperature limits to account for radiation effects.

The current heatup and cooldown curves (TS Figures 3.4-2 and 3.4-3) are licensed through the first 36 EFPY. The heatup and cooldown curves are based on the limiting intermediate shell plate B4197-2, with a 1/4T ART of 191°F and 3/4T ART of 179°F at 36 EFPY. The revised ART at 36 EFPY for power uprate conditions on limiting intermediate shell plate B4197-2 at 1/4T and 3/4T are 190.4°F and 177.8°F respectively. The ART decreased for the power uprate at 36 EFPY, because the original fluence projection used to develop the 36 EFPY pressure-temperature limits in 1998 was conservative relative to the updated power uprate fluence projection completed in 2010. Therefore, the current heatup and cooldown curves remain valid. The MUR power uprate conditions are bounded by the current licensing basis at 36 EFPY for pressure-temperature limits. No further evaluation is required until the pressure-temperature limits are updated for operation beyond 36 EFPY.

IV.1.C.iv Low Temperature Overpressure Protection

The LTOP system provides RCS pressure relief capability during relatively low temperature operation. Two pressurizer PORVs are used for automatic pressure relief during the design basis mass input and heat input transients to prevent RCS pressure from exceeding the pressure and temperature limits of 10 CFR 50, Appendix G. TS 3.4.9.4 requires LTOP system activation in Mode 4 when the temperature in any RCS cold leg is less than or equal to 325°F, and Modes 5 and 6 with the reactor vessel head on. The PORV LTOP setpoints are shown in TS Figure 3.4-4.

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ASME B&PV Code, Section XI, Appendix G, Section G-2215 requires an effective LTOP system at temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal less than RT_{NDT} + 50°F, whichever is greater. LTOP systems shall limit the maximum reactor vessel pressure to 100% of the pressure-temperature limits. The LTOP required setpoint for the limiting intermediate shell plate B4197-2 is 254.4°F, which is less than the current TS enable temperature of 325°F.
A parametric analysis was previously performed on the LTOP system design basis mass input and heat input transients. The design mass input transient is the injection of water from the start of one charging/safety injection pump into a water-solid RCS while at shutdown conditions. The design basis heat input transient is the startup of an idle RCP, with all other loops inactive and the SG secondary side 50°F hotter than the RCS primary side. The peak RHR relief valve inlet pressure was also calculated for the design basis mass input and heat input transients.
The critical analysis input parameters for the LTOP parametric study are:

- RCS volume
- SG characteristics (heat transfer rate, initial water mass, temperature differential between primary and secondary side)
- RCP startup characteristics

In addition to these parameters, the mass input flow rates, heat input initial RCS/SG temperatures, PORV stroke time and relief rate (valve C_v) impact the peak and minimum pressures during the design basis mass input and heat input transients. These critical input parameters do not change for the power uprate.

The RHR analysis calculated the peak RHR valve inlet pressure for the design basis mass input transient (up to a flow rate of 700 gpm) and heat input transient (maximum RCS temperatures of 300°F). The power uprate does not impact the key analysis inputs used for the RHR analysis, and the calculated peak RHR valve inlet pressures remain valid at power uprate conditions.

The results and conclusions of the LTOP system design basis mass and heat injection parametric analyses and RHR relief valve analysis remain valid at power uprate conditions. The design inputs to the LTOP PORV setpoint analysis were reviewed for applicability. These design inputs are unaffected by the power uprate. Therefore, the LTOP PORV setpoints determination analysis of record remains applicable at MUR power uprate conditions.

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IV.1.C.v Effect on Upper Shelf Energy Calculation

Upper shelf toughness was evaluated to ensure compliance with 10 CFR 50, Appendix G. If the limiting reactor vessel beltline material's Charpy upper shelf energy is projected to fall below 50 ft-lb, an equivalent margins assessment must be performed. The limiting vessel beltline material is the intermediate shell plate B4197-2.

Reactor vessel integrity may be affected by changes in temperatures and pressures resulting from the power uprate. An evaluation was performed to assess the power uprate impact on the Charpy upper shelf energy values for all reactor vessel beltline and extended beltline materials (i.e., locations where the 55 EFPY fluence > 1.0E17 n/cm^2) in the reactor vessel. The decrease in Charpy shelf energy due to 55 EFPY fluence at the 1/4T location was calculated using procedures specified in RG 1.99, Revision 2, Positions 1.1 and 2.1 (Reference IV-12). Based on the current analysis, all reactor vessel materials have Charpy upper shelf energy greater than the 50 ft-lb acceptance criteria of 10 CFR 50, Appendix G through 55 EFPY, including the MUR power uprate. Therefore, no further evaluation is required.

IV.1.C.vi Surveillance Capsule Withdrawal Schedule

The reactor vessel material surveillance program is used to monitor the changes in reactor vessel materials due to neutron exposure to verify that the shift has remained within acceptable limits. Test specimens representative of various reactor vessel plates and weldments were initially installed inside the reactor vessel inside capsules which are located such that the amount of neutron exposure they receive is representative of the actual reactor components.

A withdrawal schedule has been established to periodically remove surveillance capsules from the reactor vessel, to monitor the reactor vessel materials under actual operating conditions. The schedules meet the requirements of ASTM E185-82 (Reference IV-13) and 10 CFR 50, Appendix H. The reactor vessel surveillance program tests the reactor vessel surveillance capsule test specimens to monitor for neutron irradiation-induced embrittlement in base metals (plate or forgings) and welds in the beltline region of the low-alloy steel reactor vessel. There are six surveillance capsules, each with mechanical test specimens, Charpy V-Notch specimens, dosimetry, and thermal monitors. The program monitors fracture toughness of beltline materials indirectly through measurement of the impact energy of Charpy V-Notch specimens. There are two sets of specimens, one made from representative limiting beltline material (intermediate shell plate B4197-2) and the other from a non-limiting beltline circumferential weld

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(intermediate shell to lower shell weld 5P6771). The power uprate EOL peak base metal fluence at 55 EFPY is projected as 6.88E19 n/cm^2.

Three in-vessel surveillance capsules have been withdrawn to date and tested.

- Capsule U (1 RFO - fluence 0.55E19 n/cm^2)
- Capsule V (3 EFPY - fluence 1.32E19 n/cm^2)
- Capsule X (9 EFPY - fluence 3.25E19 n/cm^2)

Capsule W was withdrawn during the Fall 2010 outage (end of Cycle 16), with a projected fluence of 6.89 E19 n/cm^2. The Capsule W analysis will be used to optimize the neutron exposure and withdrawal schedule for the remaining two capsules (Y and Z), to obtain meaningful metallurgical data. HNP will adjust the withdrawal schedule for one of the remaining capsules based on the Capsule W analysis, so that capsule fluence will not exceed twice the maximum vessel fluence per ASTM E185-82. If the last capsule's projected fluence value is excessive, it will either be relocated to a lower neutron flux position, or withdrawn for possible future testing or reinsertion. One capsule will continue to be irradiated to at least the projected reactor vessel fluence at 80 years of operation, withdrawn and tested consistent with the industry PWR coordinated surveillance plan currently under development.

IV.1.D Codes of Record

Table IV-2
Codes of Record

Component	Code	Code Class	Edition and Addenda
Reactor Vessel	ASME III	1	1971 Edition through Winter 1971
CRDM	ASME III	1	1974 Edition through Summer 1974
Steam Generator			
Tube Side	ASME III	1	1971 Edition through Summer 1972 (1)
Shell Side	ASME III	1	1971 Edition through Summer 1972 (1)
Pressurizer	ASME III	1	1971 Edition through Summer 1972
Reactor Coolant System			
Piping	ASME III	1	1977 Edition through Winter 1979
Supports	ASME III	1	1974 Edition through Winter 1974
Surge piping	ASME III	1	1986 Edition
Surge line nozzle (pressurizer and reactor coolant loop hot leg)	ASME III	1	1989 Edition

Reactor coolant pump	ASME III	1	1971 Edition through Summer 1972
Class 1 Auxiliary Piping	ASME III	1	1971 Edition through Summer 1973
Main Steam System			
Piping (All main steam piping from the SGs up to and including the MSIV. Also, all branch lines from safety class piping to the first normally closed shutoff valve on the branch line.	ASME III	2	
Piping (Main steam piping downstream of the MSIV up to and including the last seismic restraint in the Turbine Building. This includes the atmospheric dump piping.)	(2)	(2)	

Notes:
1. Plus Appendix F from the 1986 Edition and material properties from the 1986 Edition for materials not available in the 1971 Code (materials other than Inconel 690). For Inconel 690, the material properties are from Code Cases N-20-3, N-747-1 and from the 1992 Code.
2. ASME Section III, Class 3 (design only) and ANSI B31.1 (fabrication only).

There are no changes to the codes of record listed above in Table IV-2.

IV.1.E Changes to Component Inspection and Testing Programs

IV.1.E.i Inservice Testing Program

10 CFR 50.55a(f), *Inservice Testing Requirements,* requires the development and implementation of an IST Program. HNP has developed and is implementing an IST Program for pumps and valves per the applicable requirements. TS 6.8.4.m describes the surveillance requirements that apply to the inservice testing of ASME B&PV Code Class 1, 2, and 3 components.

The applicable system analyses were reviewed to determine if the power uprate would impact the existing IST Program. There are no significant changes to the design basis requirements that would affect component performance, test acceptance criteria, or reference values. Test frequencies are independent of core power level and are unaffected. Therefore, the existing IST Program requirements are not impacted by the MUR power uprate.

IV.1.E.ii Inservice Inspection Program

10 CFR 50.55a(g), *Inservice Inspection Requirements*, requires the development and implementation of an ISI Program. The applicable program requirements are specified in ASME B&PV Code, Section XI. The HNP ISI Program includes containment inservice inspection, risk informed inservice inspection, augmented inservice inspections, and system pressure testing requirements imposed on or committed to by HNP. HNP has adopted EPRI Topical Report TR-112657, Rev.B-A (Reference IV-8) methodology, supplemented by Code Case N-578-1, for implementing risk informed inservice inspections. The risk informed inservice inspection program implementation is in accordance with Relief Request 13R-02. HNP has also adopted EPRI Topical Report TR-1006937, Rev.0-A (Reference IV-9) methodology for additional guidance on adapting the risk informed inservice inspection evaluation process to break exclusion region piping, also referred to as the high-energy line break region.

The power uprate does not affect any existing system or component classifications or boundaries for any ASME B&PV Code Class 1, 2, or 3 systems or components. There is no affect on the inspection requirements for any ASME B&PV Code Class 1, 2, or 3 components and their supports. The power uprate does not require the re-rating of any system components, and hydrostatic testing requirements remain applicable. There are no new or revised high-energy lines created, so additional augmented inspection is not required. Therefore, the existing ISI Program requirements are not impacted by the MUR power uprate.

IV.1.E.iii Erosion/Corrosion Program

HNP has established and maintains a FAC Program per GL 89-08 (Reference IV-10). The FAC Program is based on the guidance of EPRI NSAC-202L-R3 (Reference IV-11). This program provides a standardized method of identifying, inspecting, and evaluating piping systems susceptible to FAC. Program elements include: FAC susceptibility analysis and modeling, FAC inspection and evaluation, documentation and records. In general, plant piping systems are considered susceptible to FAC unless excluded. A system is considered as "in scope" for the FAC Program (i.e., susceptible to FAC), based upon some number of components within a given system being susceptible. In scope does not mean that the entire system is susceptible. EPRI NSAC-202L-R3 provides the nominal bounding conditions used as the basis for system susceptibility exclusion or predetermined plant-specific exclusion criteria. HNP uses the CHECWORKS Steam/Feedwater Application (SFA) FAC monitoring computer code to model the thermal dynamic conditions in the secondary side high energy piping systems, in order to predict and track FAC susceptible components.

The CHECWORKS SFA model was updated to incorporate the changes associated with the power uprate, as depicted on the revised PEPSE Turbine Cycle Model. An analysis was then performed to calculate the change in CHECWORKS predicted wear rates. Some FAC susceptible lines are predicted to have a decrease in wear. For those lines with an anticipated increase in wear, the analysis results are shown in Table IV-3.

Based on the reviews conducted, no additional secondary side piping has been identified as requiring monitoring under the existing FAC Program. As demonstrated through modeling and analysis, the MUR power uprate impact on FAC wear rates is not significant. Predicted remaining service life is essentially unchanged. The existing FAC Program bounds all FAC susceptible piping, and changes to current inspection scope and frequency are not necessary.

Table IV-3
Wear Rate Analysis

CHECWORKS Model	System	Average % Increase in Wear Rate	Increase in Wear Rate (mils/yr)	Average Wear Rate Post-Uprate (mils/yr)	Increase in Flow Rate (%)
Reheater Drain to HTR 5	Heater Drains	8.2	0.19	2.46	6.2
HTR 5 to Steam Generator	Feedwater	6.2	0.08	1.31	2.0
HTR 4 to Heater Drain Pump Suction	Heater Drains	5.0	0.03	0.72	4.0
Heater Drain Pump to Mixing Tee	Heater Drains	3.8	0.07	1.81	4.0
HTR 2 to HTR 3	Condensate	2.5	0.02	0.80	1.1
Moisture Separator Drain to HTR 4	Heater Drains	2.2	0.02	1.06	6.8
HTR 3 to HTR 4	Condensate	2.0	0.02	0.83	1.1
Mixing Tee to Feed Pump	Condensate	1.1	0.01	0.96	2.0
HTR 2 to HTR 1	Heater Drains	0.9	0.01	0.77	2.0
Feed Pump to HTR 5	Feedwater	0.7	0.01	1.88	2.0

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IV.1.F Impact of NRC Bulletin 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

NRC Bulletin 88-02 required actions by operating license holders of Westinghouse designed nuclear power reactors with SGs having carbon steel support plates. SGs in this category include Westinghouse models 13, 27, 44, 51, D1, D2, D3, D4 and E. These actions were required to minimize the potential for a SGTR caused by rapidly propagating fatigue cracks such as occurred at North Anna Unit 1 on July 15, 1987. That SGTR was caused by high cycle fatigue.

As previously stated, original Model D4 SGs were replaced in 2001with Model Delta-75 SGs. This was a complete SG replacement with no parts retained from the original SGs. An evaluation was performed on the potential for high cycle fatigue in unsupported SG U-bend tubes. One of the prerequisites for high cycle SG U-bend fatigue is a dented support condition at the upper plate. This condition results from corrosion product build-up associated with drilled holes in carbon steel support plates. Since the broached stainless steel support plate used in the SGs is designed to inhibit the introduction of corrosion products, the support condition (i.e., denting) necessary for high cycle fatigue cannot occur. Therefore, high cycle fatigue issues associated with unsupported inner row SG tubes is not a concern in this model SG.

IV REFERENCES

IV-1 NRC NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,* March 1995.

IV-2 WCAP-17041-P, Rev.0, *Almaraz Units 1 and 2Baffle-Former Bolt Structural Qualification,* February 26, 2009.

IV-3 Westinghouse Nuclear Safety Advisory Letter NSAL-07-3, Rev.1, *CRDM Head Adapter Loads,* August 14, 2008.

IV-4 WCAP-15678, Rev. 1, *Regulatory Guide 1.121 Analysis for the Shearon Harris Replacement Steam Generators,* June 2001.

IV-5 EPRI Material Reliability Program (MRP)-146, *Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant Branch Lines.*

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IV-6 EPRI Material Reliability Program (MRP)-146S, *Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant Branch Line - Supplemental Guidance.*

IV-7 WCAP-14550, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Shearon Harris Unit 1 Nuclear Power Plant,* December 1996.

IV-8 EPRI Topical Report TR-112657, Rev. B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure,* December 1999.

IV-9 EPRI Topical Report TR-1006937, Rev. 0-A, *Extension of the EPRI Risk Informed Inservice Inspection (RI-ISI) Methodology to Break Exclusion Region (BER) Programs,* August 2002.

IV-10 NRC Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning,* May 2, 1989.

IV-11 EPRI NSAC-202L-R3, *Recommendations for an Effective Flow-Accelerated Corrosion Program*, May 2006.

IV-12 NRC Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials,* May 1988.

IV-13 American Society for Testing and Materials (ASTM) E185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,* July 1, 1982.

IV-14 BAW-2241P, Revision 1, *Fluence and Uncertainty Methodologies,* Framatome Technologies Inc., April 1999.

IV-15 NRC Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,* March 2001.

IV-16 Letter from Stuart A. Richards (USNRC) to J. J. Kelly (B&W Owners Group Services), *Acceptance for Referencing of Licensing Topical Report BAW-2241P, Revision 1, Fluence and Uncertainty Methodologies,* TAC No. M98962, April 5, 2000.

IV-17 Letter from Cornelius J. Gannon, Jr. (Progress Energy Carolinas) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant, Unit No. 1, Docket No. 50-400/License No. NPF-63 Application for Renewal of Operating License,* HNP 06-0136, November 14, 2006.

IV-18 WCAP-16353-P, Revision 2, *Harris Nuclear Plant Fatigue Evaluation for License Renewal,* July 2010.

IV-19 NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Design Curves to Selected Nuclear Power Plant Components,* March 1995.

V. ELECTRICAL EQUIPMENT DESIGN

1. A discussion of the effect of the power uprate on electrical equipment. For equipment that is bounded by the existing analysis of record, the discussion should cover the type of confirmatory information identified under Section II above. For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items:

   A. emergency diesel generators

   B. station blackout equipment

   C. environmental qualification of electrical equipment

   D. grid stability

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RESPONSE TO V. ELECTRICAL EQUIPMENT DESIGN

V.1.A Emergency Diesel Generators

The EDG system provides a safety-related emergency source of AC power for the engineered safety systems and selected BOP emergency loads, in the event that the normal AC power is interrupted. HNP has two EDGs dedicated to the safety-related, redundant electrical buses.

The electrical loads that changed as a result of the power uprate are not fed from the EDG system. There is no increase to the emergency bus loads supported by the EDGs. The existing 703 kW EDG loading margin is maintained. Therefore, the EDG system is not affected by the MUR power uprate.

V.1.B Station Blackout Program

10 CFR 50.63 requires each light water cooled nuclear power plant to withstand and recover from a loss of all AC power, referred to as Station Blackout (SBO). The HNP coping duration is four hours. This coping duration is based on an evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability. The evaluation was completed per NUMARC 87-00 (Reference V-1) and RG 1.155. The power uprate was evaluated for impact on the alternate AC power source, condensate storage tank inventory, Class 1E battery capacity, ventilation, compressed air, containment isolation, and reactor coolant inventory.

V.1.B.i Alternate AC Power Source

HNP is an AC-independent plant and has been assessed for its ability to cope with SBO for four hours without AC power. As an AC-independent plant, HNP relies on the DC systems for the necessary coping power and decay heat generated steam to operate the auxiliary feed water system to cool the reactor coolant system. There are no additional loads as a result of the MUR power uprate. The SBO requirements remain bounded and unchanged for the required SBO coping power sources and supplies.

V.1.B.ii Condensate Storage Tank Inventory

The condensate storage tank has adequate inventory to maintain the plant in hot standby for six hours, followed by a six hour cooldown to residual heat removal conditions. The condensate storage tank inventory analysis was based on decay heat corresponding to 2958 MWt or 102% of 2900 MWt. This analysis remains valid at MUR power uprate conditions and will continue to bound SBO requirements. The SBO requirements remain bounded by the condensate storage tank design bases.

V.1.B.iii Class 1E Battery Capacity

HNP has two Class 1E battery systems with sufficient capacity, including 10% margin, to power SBO loads for four hours. The power uprate does not affect existing DC powered loads, and there are no additional DC loads during a SBO at power uprate conditions. Therefore, the Class 1E batteries are acceptable at MUR power uprate conditions.

V.1.B.iv Loss of Ventilation

Evaluations have been performed for the following plant areas containing SBO equipment: control room, inverter room, auxiliary feedwater pump room, main steam tunnel and mechanical penetration room. These areas are unaffected by the MUR power uprate. The temperature rise evaluations are not impacted and area temperatures remain within allowable limits.

V.1.B.v Compressed Air

There are no requirements for compressed air to assure containment isolation or to cope with SBO. The MUR power uprate does not require the installation of any air-operated components for SBO. The current compressed air requirements to support SBO coping events remain applicable at MUR power uprate conditions.

V.1.B.vi Containment Isolation

The power uprate does not add or remove any containment isolation valves. The ability to operate containment isolation valves and position indication capability is not related to power level. The evaluation for containment isolation at current plant conditions remains applicable at MUR power uprate conditions.

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V.1.B.vii Reactor Coolant Inventory

The power uprate has no impact on RCS inventory during a SBO event. There are no proposed plant modifications to the RCS or CVCS that would change system leakage or charging capability during an SBO. RCS pressure is unchanged. None of the design analysis assumptions regarding RCS inventory are changed by the MUR power uprate, and the analysis of record remains valid.

V.1.B.viii Emergency Diesel Generator Reliability Program

The power uprate does not affect the existing EDG reliability program as it relates to SBO. The EDG target reliability remains 0.95.

The power uprate has no impact on the current SBO coping duration of four hours. The plant's ability to cope with a postulated SBO at MUR power uprate conditions does not require configuration changes to the components and functions credited in the existing SBO coping assessments.

V.1.C Environmental Qualification of Electrical Equipment

The term EQ applies to equipment important-to-safety. The intent is to ensure this equipment remains functional during and following design basis events. The HNP EQ Program has been developed to ensure that EQ criteria are applied to electrical equipment important to safety as specified in 10 CFR 50.49, and to document the process used to demonstrate this qualification. HNP implements the 10 CFR 50.49 requirements through a commitment to NUREG-0588 (Reference V-2), Category II (i.e., equipment qualified per IEEE 323-1971 (Reference V-3)). However, HNP chose to establish qualification to NUREG-0588, Category I requirements (i.e., equipment qualified per IEEE 323-1974 (Reference V-4)) whenever possible. Nearly all of the EQ equipment in the plant that was originally qualified to NUREG-0588, Category II has been upgraded to meet the requirements of NUREG-0588, Category I along with the supporting documentation. A sound reason to the contrary for not upgrading, as discussed in Regulatory Guide 1.89 (Reference V-5), is included in Reference V-6 for the few items that have not been upgraded to NUREG-0588, Category I.

An evaluation was conducted to determine the effects of the power uprate on the EQ of electrical equipment. The evaluation concluded that the power uprate will not impact the equipment qualification. Revised operating conditions are bounded by equipment design limits and will not adversely diminish the capability of safety-related equipment in performing their intended safety function. The electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the MUR power uprate.

V.1.D Grid Stability

The Progress Energy Transmission Operations and Planning Department performed a Generator Interconnection Impact Study. This study assessed the impact on local transmission area power flows and voltages, transmission equipment short circuit withstand/interrupting capability, the transient and dynamic stability of HNP and other nearby generation and the ability of HNP to meet the large generator interconnection power factor requirements (i.e., the generator's MVAR capability). This impact study determined that there was no power-flow, short circuit, stability, or interconnection impacts on the grid transmission system. No additions or modifications to the Progress Energy transmission system facilities are required to accommodate the proposed MUR power uprate.

V.1.D.i Background

Progress Energy Transmission Planning has evaluated a proposed increase of 66 MWe net for HNP. Plant modifications are anticipated that would result in additional electrical power increases beyond that proposed by the MUR. Grid stability studies were conducted assuming that these power increases were in effect, so the results bound the MUR power uprate. Generator interconnection impact study results are from Progress Energy's internal power-flow models. The Progress Energy internal system analysis consists of an internal Progress Energy transmission system using documented transmission planning criteria for thermal, voltage, stability or short circuit. The cases include the most recent information for load, generation additions, transmission additions, interchange, and other pertinent data necessary for analysis. For systems surrounding Progress Energy, data is based on the NERC Multiregional Modeling Working Group model.

V.1.D.ii Power-flow Analysis

With respect to local transmission area power flow and voltages, the additional MWe output will have a negligible impact on transmission line loading or area voltages. Based on the study results for Summer 2010 extrapolated for future years, no thermal or voltage impacts were identified.

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V.1.D.iii Large Generator Interconnection Power Factor Requirements

In accordance with the Large Generator Interconnection Procedure (LGIP), Progress Energy Carolinas requires new generators to be capable of delivering the proposed MW generation to the Point Of Interconnection (POI) at a 0.95 lagging power factor (PF) or better. The intent is to ensure that adequate dynamic reactive resources (MVARs) also accompany the addition of real power resources (MWs) so that adequate voltage support will exist under normal, contingency and transient disturbance conditions. For MW increases of existing generation, Progress Energy Carolinas only applies this 0.95 PF lagging requirement to the uprate portion of the total generation. Therefore, for HNP, the 0.95 PF requirement would only be applied to the 66 MW uprate and then only for the incremental increases (Queue numbers) as they are deployed.

The increase in MVAR losses in the Generator Step Up Transformers are also accounted for in determining the requirement. Based on these considerations, Progress Energy Carolinas Transmission Planning determined the required nameplate MVA and PF necessary for the generator. For a 66 MW increase on Harris Unit 1, a generator nameplate rating of 1155 MVA at 0.94 PF lagging was required. This does not mean that the Harris generator is capable of delivering its entire MW output to the POI at 0.95 PF, but confirms that the added 66 MWs will not adversely impact area voltage support capability.

V.1.D.iv Stability Analysis

An evaluation was performed to assess the impact of the proposed generation uprate on system stability. Sufficient margin exists such that the replacement main generator in conjunction with the increased MWe output will not adversely impact the transient or dynamic stability of HNP or other nearby generation. Progress Energy uses a stability criterion that requires generators be capable of withstanding a close-in, two-phase to ground fault with delayed clearing without a loss of synchronism. The Progress Energy criterion complies with the NERC Reliability Standards TPL-001/002/003 requirements for stability related planning events.

In addition to grid impacts, potential impact on HNP offsite power adequacy as a result of the proposed MWe increase was considered. In accordance with NERC Reliability Standard NUC-001, Progress Energy Transmission Planning and HNP have agreed upon Nuclear Plant Interface Requirements (NPIRs) that include a minimum required switchyard voltage and plant post-trip auxiliary loading. These NPIR values remain unchanged as a result of the MWe increase. Progress Energy Transmission Planning has evaluated the impact of the proposed MWe increase, and determined that no changes are necessary to existing transmission system operating procedures in order to provide adequate voltage support to HNP. The proposed MWe increase can be accommodated by the existing transmission system, and the transmission system will continue to be capable of providing adequate switchyard voltage support to meet HNP needs as defined in the existing NPIRs.

V.1.D.v Short Circuit Analysis

A short circuit evaluation was performed to assess the power uprate impact on transmission equipment capabilities. Progress Energy Transmission Planning considered the replacement main generator's impact on HNP switchyard equipment and other transmission equipment in the local area. While the replacement main generator would increase the level of available short circuit current, sufficient margin exists such that transmission equipment is within rated capabilities. The evaluation results indicate that the interrupting capability of the transmission equipment in the surrounding area would not be adversely impacted by the proposed generation uprate.

V.1.E Onsite Power Systems

V.1.E.i AC Distribution System

The AC distribution system is the power source for the non safety-related buses and the safety-related emergency buses. The AC distribution system receives power under normal operating conditions through the unit auxiliary transformers. Under plant startup and shutdown conditions, power is supplied through the startup transformers. The AC distribution system consists of the 6.9 kV, 480 V, and 120 V systems (excluding the EDGs). The electrical changes resulting from the power uprate occur in the BOP equipment, primarily at the 6.9 kV level. The following loads were affected: main feedwater pumps, condensate pumps, condensate booster pumps, heater drain pumps and RCPs. The net increase in brake hp is small, in the range of 100-200 hp. In all cases, the new brake hp is within the rated hp for the affected motors. Since the associated overcurrent protective relay settings for large motors are based upon rated hp, no changes are required. The affected 6.9 kV motors remain the same size and no additional 6.9 kV motor loads were added, so the available fault current from these sources does not change. The 6.9 kV bus loading levels under power uprate conditions will not result in unacceptable steady-state voltages, overloads, or exceeding short circuit ratings.

The 480 V system loading levels do not change as a result of power uprate conditions, and the system will not experience unacceptable steady-state voltages, overloads, or exceeding short circuit ratings.

The 120 V system loads are not related to the power generation process and are independent of the power uprate.

Therefore, the 6.9kV, 480 V and 120 V AC electrical distribution systems are acceptable and will perform their design functions at MUR power uprate conditions.

V.1.E.ii DC Distribution System

The DC distribution system consists of 250 V and 125 V systems. A review determined that there is no increase in electrical loading on the DC power system. The DC power system is not affected and will continue to perform its design functions under power uprate conditions. The existing DC power system margins are maintained. Therefore, the DC 250 V and 125 V electrical distribution systems are acceptable at MUR power uprate conditions.

V.1.F Power Conversion Systems

V.1.F.i Main Generator

As previously stated, HNP replaced the main generator during the Fall 2010 refueling outage. The replacement main generator has a nameplate rating of 1155 MVA (based on 75 psig hydrogen pressure), 0.94 power factor lagging (0.95 leading), and 22 kV. The maximum MWe output at rated (0.94 lagging) power factor is 1085.7 MWe.

Heat balance modeling was performed using the PEPSE software, licensed by Scientech. The original HNP PEPSE model was tuned to plant operating data to produce modeled process conditions and outputs that match the current (pre-uprate) operating conditions. The heat balances were then run with a NSSS power of 2961.7 MWt (101.7% of 2900 MWt + 12.4 MWt RCP heat input).

<u>Benchmark Power Case</u>
The benchmark case provides power cycle process conditions for the current operation. It modeled operation at an NSSS power level of 2911.5 MWt, which is slightly below the current

NSSS value of 2912.4 MWt. The benchmark case had good correlation to recorded plant operating data. The gross electrical output for this case was 992 MWe.

Power Uprate Case
The 2961.7 MWt heat balances were prepared to serve as the basis for predicting actual gross plant power generation at uprated conditions. The gross electrical output for this case is 1021.8 MWe, which is 29.8 MWe greater than the benchmark gross output. This MWe increase includes anticipated plant modifications beyond those proposed by this MUR. The intent was to perform the heat balance to accommodate these anticipated changes and bound the MUR power uprate.

The gross electrical output for this case is 1021.8 MWe, which is 29.8 MWe greater than the benchmark gross output. The 29.8 MWe includes approximately 19 MWe associated with the MUR assumed by the uprate and the balance from the other upgrades. The 19 MWe allocated to the MUR represents the conservative estimate used by the Generator Interconnection Impact Study Report to assess the impacts of generator interconnection requests on the reliability of the PEC transmission system with respect to power-flow, stability, and short circuit issues.

The main generator capability curve indicates that at 1021.8 MWe, the main generator is capable of exporting approximately 430 MVAR and importing approximately 410 MVAR. However, the main generator voltage regulator minimum excitation limiter setting (currently set at approximately 125 MVAR) will limit the imported MVARs. Therefore, MUR power uprate operation remains below the replacement main generator maximum capability and is acceptable.

V.1.F.ii Isolated Phase Bus

The isolated phase bus duct connects the main generator output to the primary windings of the main transformers and unit auxiliary transformers. The isolated phase bus consists of four segments. The first segment is the main bus, which runs from the main generator output to the main transformers delta split point. This segment has a continuous current rating of 28,830 amperes and is force air cooled. The second segment runs from the delta split point to the main transformers primary side. This segment has a continuous current rating of 16,700 amperes and is force air cooled. The third segment runs from the main bus to the primary side of the unit auxiliary transformers. This segment has a continuous current rating of 1850 amperes and is self-cooled. A fourth segment connects the main bus to the potential transformers that monitor isolated phase bus voltage and sees no appreciable current.

Only the first two of the four isolated phase bus segments are affected by the power uprate; the main bus and delta bus (main transformers). The third segment is rated to carry the unit auxiliary transformers full load rating. As discussed in Section V.1.F.iv, the increase in plant auxiliary loads due to the power uprate does not overload the unit auxiliary transformers. The fourth segment connecting to the potential transformers is unaffected.

The isolated phase bus was analyzed for the main generator's output to the main transformers under normal and abnormal operating conditions. Normal conditions are defined as full real power (1021.8 MWe) with reactive power at the maximum HNP administrative limit (currently 175 MVAR lagging) and 100% nominal voltage (22 kV). Worst case abnormal operating conditions are defined as full real power (1021.8 MWe), reactive power at the main generator capability curve (lagging) for 1021.8 MWe real power level at 75 psig hydrogen and 95% nominal voltage (20.9 kV). At the HNP procedurally limited reactive power level of 175 MVAR and rated (22kV) voltage, the main generator phase current is 27,206 amperes. At 175 MVAR and 95% nominal voltage (20.9 kV), the main generator phase current is 28,638 amperes. Both of these ampere values are within the isolated phase bus main bus rating of 28,830 amperes. The isolated phase bus rating of 28,830 amperes supports generated MVARs up to 403 at 100% voltage or 212 at 95% voltage. HNP is required to provide 0.95 power factor (PF) at the POI for only the uprate portion of the total generation. In addition, the 0.95 PF requirement applies only to 100% nominal grid voltage (1.0 per unit). HNP does not have a requirement to deliver 0.95 PF at full load and 0.95% rated voltage (0.95 per unit). The LAR demonstrates that the isolated phase bus can deliver the required MVARs associated with the MUR uprate at 100% voltage. The additional values at 95% voltage are provided for information only.

HNP will document the specific ampacity requirements of the isolated phase bus to accommodate the MUR uprate and the planned HP turbine upgrade. In the event that the required ampacity is not provided by the installed isolated phase bus, the licensee commits to either upgrading the isolated phase bus to meet the required capacity or limit output to the capacity of the installed isolated phase bus.

V.1.F.iii Main (Step-up) Transformers

The main transformer bank consists of three single-phase step-up transformers that step voltage up from a nominal 22 kV at the main generator to a nominal 230 kV transmission voltage. Each transformer is rated at 336 MVA (forced oil and air) with a temperature rise of 65°C, for a total rated capacity of 1008 MVA.

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At 1021.8 MWe when the auxiliary loads are on the startup transformers, the ratings on the existing main transformers are exceeded. PEC is procuring replacement main transformers with anticipated capacity of 425 MVA with a temperature rise of 65°C, for a total rated capacity of 1275 MVA. This capacity will carry the full main generator output under all required operating conditions. PEC anticipates that these replacement main transformers will be installed prior to MUR power uprate implementation. However, if this installation is delayed, HNP would limit the uprated plant output to within the capacity of the existing main transformers until the higher capacity main transformers are installed.

V.1.F.iv Unit Auxiliary Transformers

Two three-phase unit auxiliary transformers step down voltage from a nominal 22 kV to a nominal 6.9 kV. Each is rated at 60 MVA (forced oil and air) with a 55°C temperature rise and at 67.2 MVA (forced oil and air) with a 65°C temperature rise. There are two secondary windings (x-winding and y-winding), each rated at 6.9 kV, 30 MVA (forced oil and air) with a 55°C temperature rise, and 33.6 MVA (forced oil and air) with a 65°C temperature rise. During normal operating conditions, they power the 6.9 kV switchgear and the 6.9 kV/480 V transformers, which feed the 480 V load centers and motor control centers. The 6.9 kV buses are transferred and connected directly to the secondary of the startup transformers during station startup and shutdown conditions.

The BOP electrical loads affected by the uprate increase the loading on the unit auxiliary transformers. The net increase in brake hp is small, in the range of 100-200 hp, or approximately 0.1-0.2 MVA of each unit auxiliary transformer. The unit auxiliary transformer loading is as follows:

A Train (x-winding) 2399 amperes (approximately 28.67 MVA existing loading at 6.9 kV)
B Train (x-winding) 2639 amperes (approximately 31.54 MVA existing loading at 6.9 kV)

The power uprate increases x-winding loading by approximately 0.135 MVA, which leaves margins of 4.8 MVA and 1.9 MVA for the A Train and B Train respectively.

A Train (y-winding) 1933 amperes (approximately 23.10 MVA existing loading at 6.9 kV)
B Train (y-winding) 2021 amperes (approximately 24.15 MVA existing loading at 6.9 kV)

The power uprate increase y-winding loading by approximately 0.018 MVA, which leaves margins of 10.48 MVA and 9.43 MVA for the A Train and B Train respectively.

Therefore, the unit auxiliary transformers will not be overloaded by the small MVA increase in motor loading due to the power uprate. The limiting components downstream of the unit auxiliary transformer low voltage windings (the non-segregated phase bus duct, switchgear main breaker, and switchgear horizontal bus) are rated at 3000 amperes and are capable of carrying the full rated output of these transformers.

V.1.F.v Startup Transformers

Two three-phase startup transformers step down voltage from the 230 kV switchyard to the 6.9 kV system. Each is rated at 36/48/60 MVA (oil air/forced air/forced oil and air) with a temperature rise of 55°C, and at 67.2 MVA (forced oil and air) with a temperature rise of 65°C. Each startup transformer has two secondary windings, both of which are rated at 6.9 kV, 18/24/30 MVA (oil air/forced air/forced oil and air), with a 55°C temperature rise, and at 33.6 MVA (forced oil and air) with a temperature rise of 65°C. During plant startup and shutdown, they power the 6.9 kV switchgear and the 6.9 kV/480 V transformers, which feed the 480 V load centers and motor control centers.

The BOP electrical loads affected by the uprate are the same loads as discussed above for the unit auxiliary transformers, and increase the loading on the startup transformers. Using the 33.6 MVA (forced oil and air) 65°C rise rating, the same margins exist for the startup transformers when they are supplying plant auxiliary load as for the unit auxiliary transformers discussed in Section V.1.F.iv. Therefore, the startup transformers will not be overloaded by the increase in motor loading due to the power uprate. The limiting components downstream of the startup transformer low voltage windings (the non-segregated phase bus duct, switchgear main breaker, and switchgear horizontal bus) are rated at 3000 amperes and are capable of carrying the full rated output of these transformers.

V.1.G Switchyard

The 230 kV switchyard is described in FSAR Section 8.2. The switchyard is comprised of two 230 kV buses connected to the transmission system and startup transformers through breaker-and-a-half schemes, and connected to the main transformers through a double-breaker scheme to a 230 kV tie line. The switchyard is provided with two independent 125 VDC systems to furnish the circuit breakers control power. The preferred power supply (offsite) is the 230 kV system.

Switchyard components were evaluated at power uprate conditions to ensure they were capable of performing their design functions without exceeding the limiting design parameters. Switchyard components include: 230 kV tie line between the main transformers and the switchyard, 230 kV circuit breakers, disconnects, and buses. Equipment ratings are as follows: 230 kV tie line, 3502 amperes or 1395 MVA at 230 kV; switchyard breakers and disconnects, 3000 amperes or 1195.11 MVA continuous; and switchyard buses, 1583 MVA continuous. The maximum apparent power through these components post power uprate is approximately 2940 amperes or 1108.6 MVA. Therefore, adequate positive margin exists between the maximum worst case steady-state load and switchyard equipment ratings using conservative assumptions. The existing switchyard is capable of supporting the MUR power uprate conditions.

V REFERENCES

V-1 Nuclear Management and Resources Council, NUMARC 87-00, *Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,* November 1987.

V-2 NRC NUREG-0588, Rev.1, *Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,* July 1981.

V-3 IEEE Standard 323-1971, *General Guide for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.*

V-4 IEEE Standard 323-1974, *Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.*

V-5 NRC Regulatory Guide 1.89, Rev. 0, *Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants,* November 1974.

V-6 Nuclear Generation Group Standard Procedure EGR-NGGC-0156, *Environmental Qualification of Electrical Equipment Important to Safety*, Revision 14.

VI. SYSTEM DESIGN

1. A discussion of the effect of the power uprate on major plant systems. For systems that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II above. For systems that are not bounded by existing analysis of record, a detailed

discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following systems.

A. NSSS interface systems for pressurized water reactors (PWRs) (e.g., main steam, steam dump, condensate, feedwater, auxiliary/emergency feedwater) or boiling water reactors (BWRs) (e.g., suppression pool cooling), as applicable

B. containment systems

C. safety-related cooling water systems

D. spent fuel pool storage and cooling systems

E. radioactive waste systems

F. engineered safety features (ESF) heating, ventilation and air conditioning

RESPONSE TO VI. SYSTEM DESIGN

VI.1.A Interface Systems

VI.1.A.i Main Steam System

The main steam system is described in FSAR Section 10.3. This system was evaluated to determine the impact of the power uprate. Component parameters are bounded by the original design equipment ratings. Therefore, the main steam system is acceptable at power uprate conditions.

VI.1.A.i.1 Main Steam Piping

Main steam system pressure, temperatures and velocities were evaluated. System pressures and temperatures are bounded by piping design parameters at power uprate conditions. Main steam lines have acceptable velocities that do not exceed the industry guidelines. Therefore, main steam system piping is acceptable at MUR power uprate conditions.

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VI.1.A.i.2 Main Steam Safety Valves

A total of five ASME B&PV Code MSSVs are located on each main steam line outside the containment building and upstream of the MSIVs. MSSV lift setpoints are determined by SG design pressure and the ASME B&PV Code. The SG design pressure has not changed with the power uprate, and the minimum MSSV capacity sizing criterion is met. The existing MSSV setpoints are unchanged. Main steam overpressure events (loss of external load and turbine trip) have been analyzed at 2958 MWt or 102% of 2900 MWt. The safety analysis confirms that the MSSV capacity is adequate for overpressure protection at MUR power uprate conditions. Main steam operating pressures and temperatures are lower at power uprate conditions, therefore, the MSSVs remain bounded by their design conditions.

VI.1.A.i.3 Main Steam Isolation Valves

The MSIVs provide a means to isolate a SG in the event of a downstream steam line break. This prevents the uncontrolled blowdown of more than one SG and minimizes the associated RCS cool down and containment pressure to within acceptable limits following a main steam line break. The MSIVs are required to close within five seconds of the receipt of a closure signal, against steam break flow conditions in either the forward or reverse direction. The power uprate does not significantly affect the MSIV's ability to close within the required time period. The MSIVs are capable of accommodating the required flow rate at power uprate conditions. The worst case for differential pressure increase and thrust loads are controlled by the steam line break area, SG flow restrictor throat area, valve seat bore, and no load operating pressure. Since the power uprate does not impact these variables, the design loads and associated stresses resulting from rapid valve closure do not change. Main steam operating pressures and temperatures are lower at power uprate conditions, therefore, the MSIVs remain bounded by their design conditions and are adequately designed for uprate conditions. The MSIV bypass valves are unaffected by the power uprate.

VI.1.A.i.4 Moisture Separator Reheaters

There are two MSRs to increase the quality and enthalpy of the steam exiting the high pressure main turbine. Shell and tube side pressures and temperatures remain bounded by the MSR design conditions at power uprate conditions. Flow rates through the MSR shells are within their existing performance capacity as specified in the associated vendor manual. The tube side flow rates increase by an insignificant 0.24% from current conditions. MSR flow rates are not a limiting design parameter based on bounding MSR design pressures and temperatures. The 0.24% increase in steam mass flow rate in the primary steam lines to the MSRs is not expected to impact the MSR valves in these lines. The MSR safety relief valves have a maximum design pressure and temperature that bounds the uprate conditions. The increase in steam flow remains bounded by the calculated MSR safety valves design capacity. Therefore, the MSRs are acceptable at MUR power uprate conditions.

VI.1.A.ii Steam Dump

VI.1.A.ii.1 Steam Generator PORVs

There are three SG PORVs, one on each main steam line. The SG PORVs are located upstream of the MSIVs and adjacent to the MSSVs. The SG PORVs automatically modulate open and exhaust to the atmosphere whenever the steam line pressure exceeds a predetermined setpoint. This minimizes safety valve lifting during steam pressure transients. The set pressure for these operations is between 0-load steam pressure and the setpoint of the lowest set MSSV. Since neither of these pressures change for the proposed range of NSSS operating parameters, the SG PORV setpoint is unchanged.

The primary function of the SG PORVs is to remove NSSS heat when the MSIV is closed. These valves are also used to control plant cooldown by discharging steam to the atmosphere when the condenser, the condenser circulating water pumps, or steam dump system are not available. There is no change is SG PORV function associated with the power uprate. The installed capacity satisfies the minimum flow criteria, which is based on 2958 MWt or 102% of 2900 MWt. The original design basis, in terms of plant cooldown capability, can still be achieved over the full range of power uprate NSSS design parameters. Main steam operating pressures and temperatures are lower at uprate conditions, therefore, the SG PORVs remain bounded by their design conditions and are adequate for operation at MUR power uprate conditions.

VI.1.A.ii.2 Steam Dump System

The steam dump system provides an artificial load by dumping excess steam to the atmosphere via eight steam dump atmospheric valves, directly to the condenser via six condenser dump valves, or a combination of the two. HNP was originally designed to accommodate 100% electrical load rejection. Accordingly, the steam dump system was designed with a capacity of 70% rated full-load steam flow. This is no longer a design basis requirement, as specified in

FSAR Section 10.4.4.1. The current analyzed design basis is a maximum electrical load rejection of 50% of plant rated electrical load. A 50% electrical load rejection without reactor trip requires a steam dump system with a capacity equal to 40% of rated full-load steam flow.

A steam dump system hydraulic analysis was performed to determine steam dump system capacity at uprated conditions. This analysis concluded that for the proposed range of NSSS design parameters, the minimum steam dump system capacity with one inoperable valve would be approximately 61%. This minimum capacity exceeds the minimum steam dump system sizing requirement for the 50% electrical load rejection. The steam dump control system has two controllers, loss of load and turbine trip. Both of these transients were analyzed at 2970.6 MWt or 102% of 2912.4 MWt. The analyses showed acceptable steam dump system stability for both the loss of load and turbine trip control modes. The steam dump system piping has design pressures and temperatures that bound the maximum uprate pressure and temperature. Maximum steam dump atmospheric valve and condenser dump valve closing time and opening time is unaffected. Therefore, the steam dump system is acceptable for operation at MUR power uprate conditions.

VI.1.A.iii Extraction Steam System

The ESS heats the condensate and feedwater at various stages prior to the SGs, and provides the normal steam supply to the auxiliary steam system. This system was evaluated at power uprate conditions. The ESS piping is acceptable for uprate pressure and temperature conditions. Flow velocities do not exceed the recommended velocity guidelines with one exception. The 22 inch piping inside the condenser from the LP turbine 12th stage to feedwater heaters 2-1A and 2-1B has a maximum flow velocity of 176 feet per second, which exceeds the recommended flow velocity of 167 feet per second. This section of piping is included in the HNP FAC Program and will be monitored to ensure minimum wall thickness is maintained. The steam side of each feedwater heater is bound by its design parameters. Reverse current valves remain bounded by their design pressures and temperatures. ESS expansion joints are bound by their design pressures, temperatures, and flow rates. Drain traps are acceptable because the power uprate has a negligible impact on the quantity of condensate removed from the ESS. Therefore, the ESS is acceptable at MUR power uprate conditions.

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VI.1.A.iv Condensate and Main Feedwater Systems

The condensate and main feedwater systems are described in FSAR Section 10.4.7. These systems were evaluated to determine the impact of the power uprate.

VI.1.A.iv.1 Condensate System

There are two 50% capacity condensate pumps discharging to two 50% variable speed condensate booster pumps. Condensate booster pumps discharge through low pressure feedwater heaters to the suction of the main feedwater pumps. Two 50% heater drain pumps take suction from 4 (A and B) feedwater heater and discharge to the main feedwater pump suction.

The power uprate results in increased condensate flow of approximately 1.9%. Adequate condensate pump and condensate booster pump net positive suction head is available at normal uprate conditions. Piping pressures and temperatures are bounded by their design limiting values. Some condensate system piping flow velocities exceed the recommended limits at power uprate conditions. The lines that have operating temperatures < 200°F do not significantly impact the associated piping and are out of scope for the FAC Program. The remaining lines are included in the HNP FAC Program and will be monitored to ensure minimum wall thickness is maintained. Therefore, the condensate system will perform its design basis function adequately and is acceptable at MUR power uprate conditions.

VI.1.A.iv.2 Main Feedwater System

There are two motor-driven main feedwater pumps. These pumps are constant speed, so feedwater flow is controlled by the feedwater regulating valves on the pump discharge.

The power uprate results in increased feedwater flow of approximately 1.9%. Adequate main feedwater pump net positive suction head is available at uprate conditions. The increase in extraction steam flow through the feedwater heaters results in a small increase in feedwater temperature entering the SG. Main feedwater isolation valves, feedwater regulating valves and feedwater bypass control valves provide a containment isolation function. To accomplish this function, these valves must be capable of fast closure following receipt of any feedwater isolation signal. Conservative assumptions were used in the analysis and these assumptions are not impacted by the power uprate, so the design loads and associated stresses resulting from rapid closure of these valves do not change. Feedwater regulating valve lift at uprated conditions remains within the Westinghouse guidelines. Piping pressures and temperatures are bounded by their design limiting values. Some main feedwater system piping flow velocities exceed the recommended limits at current and power uprate conditions. These lines are included in the HNP FAC Program and will be monitored to ensure minimum wall thickness is maintained. Therefore, the main feedwater system will perform its design function and is acceptable at MUR power uprate conditions.

VI.1.A.v Feedwater Heaters

There are two parallel trains of feedwater heaters. Each train consists of five feedwater heaters. Low pressure feedwater heaters 1-4 are on the suction side of the main feedwater pumps, and high pressure feedwater heater 5 is on the discharge side of the main feedwater pumps.

The feedwater heaters were evaluated at power uprate conditions. The shell side operating pressures and temperature at uprate conditions are lower than the respective design rated values. Tube side operating pressures at uprate conditions are bound by the design pressures. The tube side design temperature will be at saturated steam conditions corresponding to the shell side design pressure. Tube side operating temperature at uprate conditions is bound by the shell side saturation temperatures. Tube side velocities are within the recommended limits. Some of the feedwater heater nozzles exceed their recommended velocities. However, these nozzles are monitored by the FAC Program for wear. Therefore, the feedwater heaters are acceptable at MUR power uprate conditions.

VI.1.A.vi Feedwater Heater and Moisture Separator Reheater Vents and Drains

During normal operation, condensate from the shell side of low pressure feedwater heaters 2 and 3 are cascaded to the shell side of the next lower pressure feedwater heater. Feedwater heater 1 drains to the main condenser. The high pressure feedwater heater 5 drains, main steam drain tank and MSR drain tank are drained to feedwater heater 4. Feedwater heater 4 drains are initially diverted to the condenser through dump valves until the heater drain pumps are started and flow is directed to the main feedwater pump suction. The operational feedwater heater vents are equipped with flow limiting orifices that are open continuously. Start-up vents are opened briefly when the feedwater heater is placed in service.

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Feedwater heaters and moisture separator reheater vents and drains were evaluated at power uprate conditions. The feedwater heaters and moisture separator reheater vents and drains are adequately designed to operate at power uprate conditions. No plant modifications are required.

VI.1.A.vii Auxiliary Feedwater System

The AFW system design basis is described in FSAR Section 10.4.9. The AFW system serves as a backup for supplying feedwater to the SGs when the main feedwater system is not available. The system includes two motor driven pumps and one turbine driven pump discharging into a common header that supplies three independent lines, one to each SG. Each pump takes suction through a common header. The CST is the normal supply source, with the emergency SW system as an alternate source when the CST is depleted. The AFW system analyses are based on a core thermal power level of 2958 MWt or 102% of 2900 MWt. The analyzed core power level remains conservative and bounds the power uprate. AFW system maximum operating temperature and pressure remain essentially unchanged. Piping and component temperature and pressure design parameters bound power uprate operating conditions. The AFW system flow requirements specified in the analysis of record are bounding. The AFW system has the capacity to provide adequate flow under transient and accident conditions. There are no changes in AFW system minimum flow requirements, and no proposed changes to AFW pump design or operation. Since no changes are being made to the pump design, the brake hp requirements are unaffected. No AFW system modifications are required to support the MUR power uprate.
The CST inventory requirement is based on transitioning the plant from full power to hot standby (Mode 3), six hours in hot standby followed by a six hour cooldown to RHR entry conditions, in the event of a LOOP. The maximum required CST volume is 211,695 gallons for the design scenario. The current analysis of record uses a core power of 2958 MWt or 102% of 2900 MWt. Therefore, core power remains conservative and bounds the power uprate. The TS minimum CST volume requirement of 270,000 gallons ensures that the usable volume bounds the maximum CST volume requirement. Therefore, the AFW system is acceptable at MUR power uprate conditions.

VI.1.B Containment Systems

The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and to limit the temperature excursion to less than the environmental qualification acceptance limits.

VI.1.B.i Containment Spray

The CS system is described in FSAR Section 6.2.2. The CS system operates in conjunction with the containment cooling system to remove heat from the containment environment, thereby limiting peak containment pressure and temperature to within the containment design limits during a LOCA or MSLB, to maintain containment structural integrity. During the initial injection mode, the CS system takes water from the RWST, mixes in sodium hydroxide from the spray additive tank to assist in iodine removal and to control sump water pH, and delivers the discharge through containment spray rings. This cooling spray removes heat from the containment atmosphere. Once the RWST reaches the low-low level, the CS pumps take suction from the containment sump.

The existing containment response analyses are based on 2958 MWt or 102% of 2900 MWt. The CS system design parameters remain bounding for the power uprate. There are no new operating requirements imposed on the CS system. There is no impact on the existing design for CS system mechanical components such as pumps, valves, tanks, piping, spray nozzles, venturis, and educators. Plant modifications are not required. Therefore, the CS system is acceptable for operation at MUR power uprate conditions.

VI.1.B.ii Containment Air Cooling

The containment cooling system is described in FSAR Section 6.2.2. It provides general area cooling and direct cooling to critical components. The system consists of four safety-related containment fan coolers (operating during normal and post accident conditions), three non safety-related containment fan coil units providing area cooling to the RCPs (operating during normal conditions only), two safety-related primary shield cooling fans supplying the annular clearance between the reactor vessel and primary shield wall, two safety-related reactor support cooling fans supplying air to the reactor vessel supports and the annular space between the reactor coolant legs and sleeves through the primary shield, and CRDM ventilation. The CRDM ventilation system is discussed in Section VI.1.B.iii. The containment ventilation system is discussed in Section VI.1.F.iii.

Containment cooling is designed to limit containment temperature to a maximum of 120°F under normal operating conditions, and in conjunction with containment spray remove heat produced during a LOCA or MSLB. Primary shield cooling minimizes concrete dehydration. Reactor supports cooling limits thermal expansion of the reactor vessel supporting steelwork.

NSSS equipment heat load changes were analyzed at power uprate conditions. The NSSS components inside containment that could potentially operate at higher temperatures are the RCS hot and cold leg piping, RCPs, reactor vessel, SGs, and the CRDMs. T_{hot} is the only NSSS design temperature that increases due to the power uprate, so only the NSSS component heat losses associated with that temperature will contribute to an increase in the containment heat load. The primary NSSS components associated with a T_{hot} design temperature are the hot leg piping and the CRDMs. The HNP reactor vessel upper head region operates at T_{cold} rather than T_{hot}, so the only NSSS component that could increase the overall containment heat load is the RCS hot leg piping. The decrease in T_{cold} more than offsets the increase in T_{hot} due to the larger length of associated piping, resulting in no change to the containment heat load from the RCS piping. The change in coolant density is negligible, therefore the resulting change in RCP motor heat loss due to the change in T_{cold} is insignificant. The overall heat losses from the NSSS will not increase from their current values due to the power uprate. The net effect on the containment heat load is negligible.

Normal and emergency electrical loads within containment are not affected by the power uprate. Heat loads from electrical components such as cable trays, lighting panels, motors and lighting will not change.

Therefore, the containment cooling system capabilities remain acceptable at MUR power uprate conditions.

VI.1.B.iii CRDM Ventilation

The CRDM ventilation system provides heat removal from the CRDM electro-magnetic coil windings. The system was evaluated at power uprate conditions to demonstrate that the electro-magnetic coils design temperature of 392°F was not exceeded.

The coil temperature after 15 minutes of stepping is the limiting case for maximum coil temperature. The heat load from the CRDMs to the cooling air depends on the fluid temperature underneath the reactor vessel head. The head temperatures trend with the reactor vessel inlet temperature (T_{cold}). The vessel inlet temperature decreases from the existing temperature as a result of the power uprate. Therefore, the CRDM coil temperatures are expected to decrease slightly and consequently, the heat load to the containment will decrease slightly.

The CRDM coil operating temperatures remain below their design temperature limits at power uprate conditions. There is no significant increase in CRDM ventilation system heat load resulting from the power uprate. Therefore, the CRDM cooling system is acceptable at MUR power uprate conditions.

VI.1.C Safety-Related Cooling Water Systems

VI.1.C.i Component Cooling Water System

The CCW system is described in FSAR Section 9.2.2. CCW serves as a closed loop intermediate cooling system between the RCS and the SW system. There are three CCW pumps and two CCW heat exchangers. The third CCW pump is a swing pump that can be placed in service to replace either of the other two pumps. The CCW system provides the cooling requirements for all phases of plant operation including: startup, power operation, shutdown, refueling, and design basis accident cooldown. The CCW system design is based on a maximum SW supply temperature of 95°F.

The CCW system was evaluated to confirm that the heat removal capabilities are sufficient to satisfy the power uprate heat removal requirements during the normal plant operation, refueling, shutdown and accident cooldown conditions. The power uprate will not increase the normal system operating heat loads and will not significantly increase heat loads for refueling, shutdown and accident cooldown cases. However, revised CCW heat exchanger tube plugging requirements reduce the CCW heat exchanger effectiveness, causing a small increase in analyzed CCW operating temperatures for all cases. The increased RHR and letdown heat loads further affect the temperatures for refueling, shutdown, and accident cooldown cases. The bounding case was more rigorously analyzed to ensure that the maximum system conditions do not exceed the currently analyzed maximum conditions. The cooldown times for the normal and single-train cases increase with the revised CCW tube plugging requirements and increased RHR and letdown heat exchanger loads, but remain reasonable.

The evaluation of CCW system heat removal capabilities confirmed that at uprated conditions, cooling of the affected NSSS components during normal and post-accident operation continues to meet the applicable system functional requirements and performance criteria. All component outlet temperatures are below the CCW system design temperature of 200°F, with the exception of the gross failed fuel detector heat exchanger outlet temperature during plant shutdown at 350°F with minimum CCW flow. HNP will increase CCW flowrate to the gross failed fuel detector heat exchanger for the plant shutdown at 350°F, to ensure that the CCW outlet temperature from this component remains below the system design temperature of 200°F for all plant operating modes. There are no required CCW system modifications as a result of the power uprate. Therefore, the CCW system remains acceptable for operation at MUR power uprate conditions.

VI.1.C.ii Service Water System

The SW system is described in FSAR Section 9.2.1. It is made up of two subsystems: the normal SW system and the emergency SW system. There are two normal SW pumps and two emergency SW pumps. The normal SW system removes heat from plant auxiliary components during normal plant operation, including startup and shutdown, and transfers the heat into the cooling tower. The normal SW system also provides all cooling water requirements to the emergency SW loads during normal operation, but is not safety-related. One normal SW pump taking suction from the circulating water cooling tower basin supplies all loads during these conditions. Following an accident, the emergency SW pumps take suction from the UHS, circulate the water through the plant components required for reactor safe shutdown, and return it to the UHS.

The various systems and components cooled by the SW system were evaluated to confirm that the SW system remains capable of removing power uprate heat loads during normal, shutdown, and accident conditions. The uprate will not significantly impact the heat loads and temperatures during normal and emergency operations. There are no required SW flow changes. The SW system and component design parameters remain bounding. No system modifications are required to support the power uprate. Therefore, the SW system is acceptable for operation at MUR power uprate conditions.

VI.1.C.iii Ultimate Heat Sink

FSAR Section 9.2.5 describes the UHS. The UHS uses two alternate sources of cooling water: the auxiliary reservoir and the main reservoir. The auxiliary reservoir is the preferred source of cooling water for emergency conditions, with the main reservoir providing a backup supply. TS 3.7.5 provides limits on both reservoirs.

The auxiliary heat loads in the UHS analysis are not changing. There are no changes to the CCW and SW flow rates. The most limiting CCW temperature for the power uprate is bound by the current analyzed conditions. No system modifications are required to support the power uprate.

Therefore, the UHS analysis is bounding and the UHS remains acceptable for operation at MUR power uprate conditions.

VI.1.C.iv Residual Heat Removal System

FSAR Section 5.4.7 describes the RHR system. The maximum heat removal demand on the RHR system occurs during the plant cooldown mode of operation, when RCS sensible heat, core decay heat, and heat input from one or more RCPs must be removed. RHR cooldown performance was evaluated under power uprate conditions. The following cooldown scenarios were analyzed:

- Normal two train cooldown with staged RCP reduction and 95°F SW
- Two train cooldown with no RCPs (LOOP) and 95°F SW
- Single train cooldown with one RCP and 95°F SW
- Single train cooldown with no RCPs (LOOP) and 95°F SW
- Single train cooldown with one RCP and 90°F SW

The acceptance criteria for the RHR system functional capability require the system to cool the reactor during shutdown within the following performance specifications:

- Normal two train plant cooldown from 350°F (four hours after shutdown) to 140°F should be achieved within a reasonable amount of time after shutdown. The two train cooldown with a LOOP has no defined acceptance criteria.
- Single train cooldown from 350°F (six hours after shutdown) to 200°F should be achieved within a reasonable amount of time after shutdown, assuming either 95°F or 90°F SW.
- The most limiting single train cooldown (one RCP and 95°F SW) shall achieve cold shutdown within a reasonable amount of time after reactor shutdown.

The plant cooldown scenarios listed above were analyzed at power uprate conditions using the Westinghouse proprietary RHRCOOL code. The code models the interactive effects of the RHR system, CCW system, and SW system during plant cooldown beginning at an RCS temperature of 350°F and continuing to the specified final temperature.

The most limiting two train plant cooldown case (staged RCP reduction and 95°F SW) resulted in a maximum cooldown time of approximately 25 hours after shutdown. The most limiting single train case (one RCP and 95°F SW) resulted in a maximum cooldown time of approximately 51 hours after shutdown. These cooldown times are considered reasonable and meet the acceptance criteria. The previous hydraulic analyses of RHR flow rate during cooldown and RHR pump available net positive suction head are unaffected by the power uprate conditions and remain applicable. Therefore, the RHR system will continue to meet its design basis functional requirements and performance criteria for plant cooldown at MUR power uprate conditions.

VI.1.D Spent Fuel Pool Storage and Cooling Systems

FSAR Section 9.1.3 describes the SFP cooling and purification system. There are two 100% capacity fuel pool cooling and cleanup systems: one to service Pool A and Pool B in the South end of the fuel handling building and one to service Pool C and Pool D in the North end of the fuel handling building. SFP cooling heat exchangers are cooled by component cooling water. The clarity and purity of the fuel pool water is maintained by passing approximately 5% of the cooling system flow through a cleanup loop consisting of two filters and a demineralizer.

SFP bounding heat loads and system performance parameters are not affected by the power uprate. The amount of radioactive fission and corrosion products added to the SFP is not affected. The SFP cooling and purification system has the capability to maintain SFP radionuclide concentrations and water chemistry within acceptable limits at uprate conditions. SFP evaporative and leakage losses are not affected. The existing makeup capabilities can maintain the SFP at its required water level. Therefore, the SFP cooling and purification system is acceptable at MUR power uprate conditions.

VI.1.E Radioactive Waste Systems

VI.1.E.i Gaseous Waste

The gaseous waste processing system and its various subsystems and components were evaluated for the power uprate. Gaseous waste processing system functions and the volume of waste gas processed are unaffected. Existing system and equipment design margins are maintained, because system flow rates, gaseous inventories, and process conditions remain within the original system design parameters. The gaseous waste processing system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

VI.1.E.ii Liquid Waste

The liquid waste processing system and its various subsystems and components were evaluated for the power uprate. Liquid waste processing system functions and the volume of liquid waste processed are unaffected. The concentration of radionuclides in the liquid is expected to increase by a small amount. This increase does not significantly impact system operation. Existing system and equipment design margins are maintained, because system flow rates, liquid inventories, and process conditions remain within the original system design parameters. The liquid waste processing system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

VI.1.E.iii Solid Waste

The solid waste processing system is not currently in-service and most of the original processing equipment has been abandoned. Abandoning this equipment does not prevent the solid waste processing system from being reinstated in the future, if required. Solid wastes are presently processed by a vendor-supplied and designed solid radwaste processing system. These wastes are packaged by outside service vendors in high integrity containers, liners, and drums as required for shipment to an offsite disposal facility. There is no long-term storage of packaged solid waste liners, containers, or drums at the plant site. Solid waste volumes are not significantly affected by the power uprate. The solid waste processing system, currently not in operation, and the solid radwaste processing system are capable of processing the expected levels of uprated solid waste, because the quantities of solid waste and the processing conditions are not significantly affected and remain within system operating margins. The solid waste processing system and solid radwaste processing system remain adequate and are acceptable at MUR power uprate conditions.

VI.1.E.iv Steam Generator Blowdown

The required SG blowdown flow rates during plant operation are based on chemistry control and tubesheet sweep necessary to control solids buildup. The blowdown volumetric flow rate is not increasing at power uprate conditions, so associated system flow velocities will not increase. Blowdown system operating temperatures and pressures will decrease and remain bounded by the existing design parameters. The small decrease in SG pressure, approximately 7 psi, may cause blowdown flow control valves to open slightly to accommodate the same flow rate into the flash tank. The SG blowdown system piping and components were evaluated for the affects of

FAC at uprate conditions. System velocities are not increasing, so the wear rate due to FAC does not increase. Therefore, the SG blowdown system will continue to meet system design requirements at MUR power uprate conditions.

VI.1.F Engineered Safety Features (ESF) Heating, Ventilation, and Air Conditioning

The heating, ventilation and air conditioning systems are adequately designed to operate at power uprate conditions. Pressure, temperature, and relative humidity operating requirements are bounded by current system capabilities. Modifications are not required to support the power uprate. The existing limiting case heat loads remaining bounding at MUR power uprate conditions.

VI.1.F.i Control Room Ventilation System

The control room ventilation system is described in FSAR Section 9.4.1. The system supplies conditioned air to the control room envelope, consisting of the control room, office area, relay and termination cabinet rooms, kitchen and sanitary facilities, and the CCW surge tank room. The system is designed to maintain an average temperature of 75°F and maximum relative humidity of 50% during all modes of operation.
Control room air conditioning system design is based on heat loads from the following:

- Heat transmitted through the building structure due to temperature differences and solar heat gain
- Heat dissipated from electrical equipment, controls, lights, and people within the control room, panel room, and other areas within the envelope
- Outside air heat admitted into the control room envelope, as required, for pressurization and ventilation

These heat loads were evaluated at power uprate conditions. Heat loads from electrical components such as panels, switchgear, or cabinets do not change. There are no personnel or lighting changes required. Normal and emergency electrical loads within the control room are unaffected. Control room humidity sources (personnel and outside air) are not changing. The power uprate does not affect the control room pressurization or filtration parameters required for radioactivity exclusion. Therefore, the control room ventilation system capability to provide appropriate temperature, humidity and air quality for equipment and personnel during normal and emergency conditions remains adequate and acceptable at MUR power uprate conditions.

VI.1.F.ii ESF Ventilation Systems

FSAR Section 9.4.5 describes the ESF ventilation systems. Dedicated ventilation subsystems maintain suitable operating environments for ESF equipment. The ESF ventilation systems are comprised of the following:

- RAB ESF equipment cooling system
- RAB switchgear rooms ventilation system
- RAB electrical equipment protection room ventilation system
- Ventilation systems for the fuel oil transfer pump house, EDG building, ESW intake structure ventilation system, and spent fuel pump room. The spent fuel pump room is discussed in Section VI.1.F.iv.

The ESF ventilation systems are designed to remove thermal loads attributable to electrical and mechanical components and/or heat transferred through building structures due to temperature differences and solar radiation, and heat admitted through air intakes. The ESF equipment cooling system capability is not affected because there is no increase in piping system heat loads or electrical heat loads at uprate conditions. The heat loads associated with the RAB switchgear rooms and RAB electrical equipment protection room are not changing and those ventilation systems remain adequate at uprate conditions. EDG building heat loads, including the fuel oil transfer pump house are not changing and the associated ventilation systems remain adequate at uprate conditions. ESW intake structure heat loads include environmental and those attributable to electrical and mechanical components. These heat loads are not changing and the ESW intake structure ventilation system remains adequate at uprate conditions. Therefore, the ESF ventilation systems remain adequate at MUR power uprate conditions.

VI.1.F.iii Containment Ventilation System

FSAR Section 9.4.7 describes the containment ventilation system. The purpose is to provide general area airborne radioactivity removal, containment atmosphere purge exhaust, containment purging prior to personnel entry, and containment vacuum relief if it exceeds established limits. The system consists of four separate sub-systems: airborne radioactivity removal, normal containment purge makeup and exhaust, containment pre-entry purge makeup and exhaust, and containment vacuum relief.

The power uprate does not require modifications that would change containment air volume. Therefore, the functions to maintain the containment at a slight vacuum, relieve excessive containment vacuum, and purge the containment atmosphere prior to personnel entry are not impacted by the power uprate. Airborne radioactivity at uprate conditions has been analyzed and containment radiological loading does not increase, so airborne radioactivity removal remains adequate. Therefore, the containment ventilation system capabilities at MUR power uprate conditions are acceptable.

The containment cooling system is discussed in Section VI.1.B.ii.

VI.1.F.iv Fuel Handling Building Ventilation System

The FHB ventilation system is described in FSAR Section 9.4.2. The purpose is to maintain a FHB operating floor environment suitable for personnel comfort and safety, minimize operator exposure to SFP water evaporation, isolate the operating floor in the event of a radioactive material release, and provide cooling for operating equipment and ventilation for areas below the operating floor. The system consists of four separate sub-systems: FHB normal operating floor air conditioning system, FHB operating floor emergency exhaust system, area below the operating floor ventilation system, and SFP pump room ventilation system.

FHB operating floor thermal loads are comprised of the following:

- Heat dissipated (sensible and latent) from the fuel pool areas, and mechanical and electrical equipment and components
- Heat transmitted through the building structure due to temperature differences or solar heat
- Outside air heat admitted into the building by the normal supply system to satisfy ventilation and exhaust requirements

The power uprate does not affect the normal or emergency electrical loads in the operating floor area. The heat loads from electrical components such as panels, switchgears, or cabinets will not change. Heat admitted into the operating floor area from adjacent areas, solar radiation, and air intakes is not affected. Because the design basis heat loads are not impacted, calculated space temperatures remain bounding and the FHB operating floor air conditioning system is acceptable at uprate conditions.

The operating floor emergency exhaust system is acceptable, because the air flow rate exhausted and amount/concentration of radioactive particles will not increase beyond design.

SFP heat loads remain bounded at power uprate conditions. Other heat loads, such as electrical equipment, transmission from adjacent areas and personnel, will not increase. Therefore, the FHB below operating floor ventilation system is adequate at uprate conditions.

The SFP pump room piping heat loads will not increase. Heat loads from mechanical equipment will not change. Electrical equipment heat loads are unchanged. Heat loads from adjoining areas are not affected. The calculated heat loads and space temperatures remain applicable and the SFP pump room ventilation system is adequate at power uprate conditions.

Therefore, the FHB ventilation system is acceptable at MUR power uprate conditions.

VI REFERENCES

None

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VII. OTHER

1. A statement confirming that the licensee has identified and evaluated operator actions that are sensitive to the power uprate, including any effects of the power uprate on the time available for operator actions.

2. A statement confirming that the licensee has identified all modifications associated with the proposed power uprate, with respect to the following aspects of plant operations that are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins:

   A. emergency and abnormal operating procedures

   B. control room controls, displays (including the safety parameter display system) and alarms

   C. the control room plant reference simulator

   D. the operator training program

3. A statement confirming licensee intent to complete the modifications identified in Item 2 above (including the training of operators), prior to implementation of the power uprate.

4. A statement confirming licensee intent to revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude should be reduced from the pre-power uprate value of 2% to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.

5. A discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review including:

   A. A discussion of the effect of the power uprate on the types or amounts of any effluents that may be released offsite and whether or not this effect is bounded by the final environmental statement and previous Environmental Assessments for the plant.

B. A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure.

RESPONSE TO VII. OTHER

VII.1 Operator Actions

Operator actions included in the safety analyses were reviewed for potential power uprate impact. Operator actions associated with the following events were reviewed:

Pipe Break Flooding in Reactor Auxiliary Building	FSAR Section 3.6A.6.4
Loss of Component Cooling Water to RCP	FSAR Section 5.4.1.3
Leakage from ECCS during Long-Term Recirculation	FSAR Section 6.3.2.5.2.2
Component Cooling System Alignment - Post LOCA	FSAR Section 9.1.3.3
Safe Shutdown Fire Analysis	FSAR Section 9.5.1.3(1)
Main Steam Line Break	FSAR Section 15.1.5
Feedwater System Pipe Break	FSAR Section 15.2.8
CVCS Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	FSAR Section 15.4.6
LOCA Outside Containment	FSAR Section 15.6.2
Steam Generator Tube Rupture	FSAR Section 15.6.3
Small Break LOCA	FSAR Section 15.6.5
Large Break LOCA	FSAR Section 15.6.5

1. FSAR Section 9.5.1.3 describes the fire hazard safe shutdown analyses. The analyses are maintained in engineering calculations.

The safety analysis reviews determined that the existing required operator actions are not affected by the power uprate. There is no reduction in time for required operator actions. No new manual operator actions were created and no existing manual actions were automated.

The power uprate is being implemented under the administrative controls of the engineering change process. Other potential impacts on operator actions and action times in plant procedures may be identified and evaluated during the plant modification impact reviews. The plant modification process ensures that impacted procedures will be revised prior to the MUR power uprate implementation.

VII.2.A Emergency and Abnormal Operating Procedures

Emergency and abnormal procedures were reviewed to determine any power uprate impact. There are no mitigating action or step changes as a result of the power uprate. However, the review identified two EOP setpoints that require revision, because these setpoints were developed using full power RCS hot leg temperature and the full power RCS hot leg temperature changed with the power uprate. These EOP setpoints will be revised to reflect a total core power of 2958 MWt or 102% of 2900 MWt, which bounds the power uprate.

The review identified the following three AOPs that require revision:

- AOP-010 (Feedwater Malfunction), because immediate actions include specific power levels.
- AOP-035 (Main Transformer Trouble), due to the higher output after MUR implementation.
- AOP-038 (Rapid Downpower), due to rescaling the first stage pressure instruments.

The EOP and AOP changes do not significantly affect operator actions and mitigation strategies.

There are no operator action changes for shutdown risk management. The time to boil will decrease due to the power uprate, but the method of calculating the time to core boil will remain the same. Procedures will be revised with data generated with decay heats at the uprated power level.

Procedure changes and any associated operator training will be completed during MUR power uprate implementation and prior to operation above 2900 MWt.

VII.2.B Control Room Controls, Displays and Alarms

The following changes/modifications associated with the proposed power uprate affect control room controls:

- Instruments associated with turbine first stage pressure will require scaling changes for NSSS protection permissive P-13 (turbine at power) and control permissives C-5 (auto rod control), C-7A & B (load rejection), and C-20 (AMSAC arming).

The following modifications associated with the proposed power uprate affect operator displays (including the safety parameter display system (SPDS)).

- Instrument loops are affected by the power uprate (possible indicator replacement, calibration span, and/or scaling).
- Plant computer points will be added and/or changed for the revised calorimetric algorithm and the feedwater LEFM.
- No significant SPDS changes are anticipated as a result of the power uprate. Critical safety function status trees will be reviewed and revised as necessary.
- The new LEFM electronic cabinet, located in the Secondary Sampling Equipment Enclosure in the Turbine Building, is used to display feedwater flow data. The display provides system status or monitored process parameters. The display is typically used for maintenance purposes and not for control of plant operations.
- The LEFM system will provide input to the secondary calorimetric. LEFM system parameters will be displayed in the Main Control Room through an ERFIS interface. There are two system channels (A & B). The operator will be able to observe feedwater flows, pressures, and temperatures for both channels on each feedwater loop and total feedwater flow. Reactor power will be displayed in both percent and MWt for the one minute and one hour average. The 12-hour average will be displayed in MWt.

The following modifications associated with the proposed power uprate affect alarms.

- The system alerts operations personnel to LEFM trouble through main control room annunciator Computer Alarm Reactor. This annunciator alerts the operators when the system loses a plane of operation, has a channel that reaches an Alert or Fail condition, a high temperature condition, or other failures. Any LEFM condition that increases feedwater flow uncertainty requires operator attention.

VII.2.C Control Room Plant Reference Simulator

The power uprate is being implemented under the plant modification process administrative controls. As part of this process, potential simulator modifications will be identified. Simulator required changes resulting from the power uprate will be evaluated, implemented, and tested per approved procedures. Simulator fidelity will be revalidated per approved procedures. Any required simulator modifications will be completed in time to support operator training prior to MUR power uprate implementation.

VII.2.D Operator Training Program

The operator training program requires revision as a result of the power uprate. Operator training will be developed and the operations staff trained on the plant modifications, TS changes, new Relocated TS and Design Basis Requirements attachment, and procedure changes prior to MUR power uprate implementation.

VII.3 Intent to Complete Modifications

HNP will complete the modifications required to support the uprate (including operator training) prior to MUR power uprate implementation.

VII.4 Temporary Operation Above Licensed Power Level

HNP will revise the existing plant operating procedure(s) related to temporary operation above full steady-state licensed power levels, as necessary, to account for the uprated power level.

VII.5 10 CFR 51.22 Discussion

10 CFR 51.22 provides criteria for and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment.

VII.5.A Power Uprate Effect on Types or Amounts of any Effluents that may be Released Offsite

The proposed change does not involve installing new equipment or modifying existing equipment that might affect the types or amounts of effluents released offsite.

There will be no significant change in the types or significant increase in the amounts of any effluents released offsite during normal operation. The primary coolant specific activity is expected to increase by no more than the percentage increase in power level.

Gaseous and liquid radwaste effluent activity is expected to increase from current levels by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be within allowable 10 CFR 20 and 10 CFR 50, Appendix I limits per the HNP Offsite Dose Calculation Manual. The proposed changes will not result in changes to the operation or design of the gaseous or liquid waste systems and will not create any new or different radiological release pathways.

Solid radwaste effluent activity is expected to increase from current levels proportionally to the increase in long half-life coolant activity. The total long-lived activity is expected to be bounded by the percent power uprate. Changes in solid waste volume are expected to be minor.

During power uprate operation, the non-radiological effluents will continue to be within regulatory standards for offsite outfalls and internal onsite outfalls.

Therefore, the license amendment request will not result in a significant change in the types or significant increase in the amounts of effluents that may be released offsite.

VII.5.B Power Uprate Effect on Individual or Cumulative Occupational Radiation Exposure

The license amendment request does not significantly increase core power and resultant dose rates in accessible plant areas. Normal operation radiation levels will increase by approximately the percentage or core power uprate. The power uprate does not require additional radiation shielding to support normal plant operation. Individual worker exposures will be maintained within regulatory limits and ALARA by the HNP Health Physics Program, which controls access to radiation areas and maintains compliance with 10 CFR 20.

Therefore, the license amendment request does not result in a significant increase to the individual or cumulative occupational radiation exposure.

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VII.6 Programs and Generic Issues

VII.6.A Fire Protection Program

FSAR Section 9.5.1 describes the Fire Protection Program. The Fire Protection Program adopts NFPA 805 (Reference VII-1) per 10 CFR 50.48(c) and serves as the HNP method of satisfying 10 CFR 50.48(a) and General Design Criteria 3.

VII.6.A.i Fire Protection Systems

The fire protection systems consist of the following major subsystems: fire detection (including smoke detectors, heat detectors, alarms), water suppression (including fire pumps, piping, sprinkler systems, deluge systems, Halon suppression, manual fire equipment (portable fire extinguishing equipment), and fire barriers (including fire walls, fire doors, penetration seals, cable wraps, heat shields). The fire protection subsystems remain unchanged as a result of the MUR power uprate.

VII.6.A.ii Responsibilities

Plant management, supervisory and station personnel responsibilities in support of the Fire Protection Program are not impacted by the power uprate, and additional personnel are not required for station fire response.

VII.6.A.iii Administrative Controls

Topics include control of combustibles, control of ignition sources, design change control, penetration breach program, Fire Protection Program procedures and drawings, fire inspection program, and fire equipment maintenance and testing. The power uprate does not affect the established administrative controls.

VII.6.A.iv Fire Brigade

There are no changes in the fire brigade staffing, responsibilities, reporting relationships, training and qualification, or equipment requirements resulting from the power uprate.

VII.6.A.v Evaluations of Inadvertent Operation of Fire Protection Systems

The power uprate does not affect the existing evaluation conclusions for inadvertent operation of fire protection systems and they remain valid.

VII.6.B High Energy Line Break Program

The high and moderate energy break program ensures that systems or components required for safe shutdown or important to safety are not susceptible to the consequences of high and/or moderate energy pipe breaks. FSAR Section 3.6, *Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping*, and Appendix 3.6.A.2, *High Energy Pipe Break Outside Containment*, describe the high energy and moderate energy line break analysis. High energy pipe breaks are analyzed for piping systems which, during normal operating conditions, exceed 200°F and/or 275 psig. Moderate energy pipe breaks are analyzed for piping systems having both a maximum normal operating temperature of 200°F or less, and a maximum normal operating pressure of 275 psig or less.

The evaluation determined that the MUR power uprate does not result in any new or revised high or moderate energy pipe break locations. The existing high and moderate energy line break analysis is not affected. The current temperature and pressure response evaluations remain valid at power uprate conditions. As described in Section II.2.45, internal flooding resulting from HELBs and MELBs at uprate conditions does not significantly affect the current flooding analysis and it remains valid.

VII.6.C Appendix J Program

FSAR Section 6.2.6 describes the containment leakage tests. The performance based testing program includes Type A tests to measure the containment overall integrated leakage rate, Type B tests to measure local leakage rate across the pressure retaining components associated with containment penetrations, and Type C tests to measure containment isolation valve leakage rates. The Type A containment leakage tests are performed as required by 10 CFR 50, Appendix J, Option B; Type B and C tests are conducted in accordance with the original commitment to 10 CFR 50, Appendix J, Option A.

The current bounding accident inside containment is the LBLOCA. A review of the LBLOCA response analysis confirmed that the analysis was performed at 102% of 2900 MWt and remains bounding, with a corresponding peak containment pressure of 41.8 psig. Because the LBLOCA peak pressure analysis is unaffected by the power uprate, the test pressure specified in TS 6.8.4.k remains valid. There are no changes to the containment penetrations that are subject to Type B leakage tests and there are no changes to the containment isolation valves that are subject to Type C tests. No changes or modifications are required to the existing HNP Appendix J Program or procedures. Therefore, TS 6.8.4.k and the applicable HNP Appendix J Program procedures remain acceptable at MUR power uprate conditions.

VII.6.D Coatings Program

Protective coatings inside containment are used to protect equipment and structures from corrosion and radionuclide contamination. Coatings also provide wear protection during plant operation and maintenance activities. These coatings are subject to 10 CFR 50, Appendix B quality assurance requirements, because their degradation could adversely impact safety related equipment. Service Level 1 containment coatings are required to withstand design basis environmental conditions, including a LOCA. HNP protective coatings meet ANSI Standards N512, N101.2, and N101.4.

A review was conducted to determine the power uprate effect on the protective coatings used inside containment regarding suitability and stability under design basis LOCA conditions. The review considered containment pressure and temperature, radiation levels, and boric acid concentrations. The existing LOCA mass and energy releases are bounding for uprate conditions. The containment analyses are unaffected and the analyzed post-LOCA DBA containment peak pressure and temperature transients remain valid and bounding at uprate conditions for coatings qualification. The current inside containment dose estimates remain applicable. Therefore, the radiation level analysis conclusions remain valid for the containment coatings. The power uprate does not affect the existing containment spray/sump water pH. The uprate conditions are bounded by the currently analyzed pH range for Service Level 1coatings.

The post-LOCA containment pressure and temperature, integrated radiation dose, and pH range values are bounded by the data used to qualify the Service Level 1 containment coatings. Therefore, the Service Level 1 containment coatings remain qualified under MUR power uprate conditions.

VII.6.E NRC Generic Letters

VII.6.E.i GL 89-10 and GL 96-05 Motor Operated Valve Program

The NRC issued GL 89-10 (Reference VII-2) requiring licensees to develop a comprehensive program to ensure MOVs in safety-related systems would operate under design basis conditions. HNP provided the NRC with a GL 89-10 closure letter in Reference II-3. GL 96-05 (Reference VII-4) required licensees to establish a program to periodically verify that safety-related MOVs continue to be capable of performing safety functions within the current licensing basis. HNP responded to this GL in References II-5 and II-6.

An evaluation was conducted to determine the power uprate impact on MOV motors performance capabilities (e.g., voltage and ambient temperature). The power uprate will not affect the maximum differential pressure/line pressures that the system valves will experience. Therefore, the power uprate does not affect the calculations that determine the required MOV thrust and torque values. MOV actuator capability to produce the required thrust/torque is not affected. There are no changes to the MOV risk categories. The analytical methodology, testing methodology, and testing frequencies are not affected. No changes are required to the existing MOV Program. HNP remains in compliance with GL 89-10 and GL 96-05. Therefore, the existing MOV Program remains valid at MUR power uprate conditions.

VII.6.E.ii GL 95-07 Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves

The NRC issued GL 95-07 (Reference VII-7) to address potential pressure locking and thermal binding of safety-related power operated gate valves. HNP responded to this GL in References VII-8 and VII-9.

A review determined that the power uprate does not affect the pressure locking evaluations previously completed. The thrust and torque required to open the applicable valves remains less than the motor actuator capacities at uprated conditions. Valve design, valve function, and operational considerations/conditions are unaffected. New conditions were not created that would affect valve susceptibility to pressure locking or thermal binding. Therefore, the conclusions previously provided in Reference VII-9 and subsequent correspondence regarding valve pressure locking and thermal binding acceptability are not impacted by the MUR power uprate.

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VII.6.E.iii GL 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

The NRC issued GL 96-06 (Reference VII-10) to address hydrodynamic effects of water hammer and two-phase flow conditions on cooling systems serving containment air coolers and thermally induced over pressurization of isolated piping segments. HNP responded to this GL in References VII-11 and VII-12. The NRC issued a number of requests for additional information (RAI) regarding HNP's response to this GL. The RAI responses were reviewed by the NRC and subsequently approved in Reference VII-13.

ESW system operation during LOOP or LOOP/LOCA conditions is not affected. The power uprate does not modify system configuration or change system operation. The current accident analysis is performed at 102% of core power, and remains bounding. ESW system water hammer and two-phase flow analyses are not affected. There is no increase in the possibility of over pressurization of isolated segments of safety-related piping inside containment, including penetrations, due to the power uprate. There are no modifications to containment penetrations resulting from the power uprate. The conclusions in HNP's GL 96-06 responses and the associated NRC SER remain valid at MUR power uprate conditions.

VII.6.F Air Operated Valve Program

The AOV Program uses the following categories to distinguish AOVs, based on safety significance, mode of operation, and effect on plant reliability:

Category 1 - AOVs that perform an active Maintenance Rule function and have high safety significance.

Category 2 - AOVs that are safety-related and perform an active safety function and have low safety significance.

Category 3 - AOVs that are classified as Zero Tolerance for Equipment Failure or Generation Significant and not in Category 1 or 2.

An evaluation was conducted to determine the power uprate impact on AOV performance. The power uprate will not affect the maximum differential pressure/line pressures that the system valves will experience. Therefore, the calculations that determine the required AOV thrust and torque values are not affected. There is no impact on the compressed air system, so the AOV actuator capability to produce the required thrust/torque is not affected. There are no changes to the AOV risk categories. The analytical methodology, testing methodology, and testing frequencies are not affected. No changes are required to the existing AOV Program. Therefore, the existing AOV Program remains valid at MUR power uprate conditions.

VII REFERENCES

VII-1 National Fire Protection Association (NFPA) Standard 805, *Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants,* 2001 Edition.

VII-2 NRC Generic Letter 89-10, *Safety-Related Motor Operated Valve Testing and Surveillance,* June 28, 1989 and supplements.

VII-3 Letter from William R. Robinson (Carolina Power & Light Company) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Closure of NRC Generic Letter 89-10,* HNP 95-027, February 28, 1995.

VII-4 NRC Generic Letter 96-05, *Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,* September 18, 1996.

VII-5 Letter from William R. Robinson (Carolina Power & Light Company) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Generic Letter 96-05, "Periodic Verification of Design Basis Capability of Safety-Related Motor Operated Valves" 60-day response,* HNP 96-197, November 18, 1996.

VII-6 Letter from William R. Robinson (Carolina Power & Light Company) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Generic Letter 96-05, "Periodic Verification of Design Basis Capability of Safety-Related Motor Operated Valves" 180-day response,* HNP 97-032, March 14, 1997.

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VII-7 NRC Generic Letter 95-07, *Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves,* August 17, 1995.

VII-8 Letter from William R. Robinson (Carolina Power & Light Company) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Generic Letter 95-07 "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves 60-day Response Followup,"* HNP 95-096, November 3, 1995.

VII-9 Letter from William R. Robinson (Carolina Power & Light Company) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Generic Letter 95-07 "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves180-day Response Followup,"* HNP 96-019, February 13, 1996.

VII-10 NRC Generic Letter 96-06, *Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,* September 30, 1996.

VII-11 Letter from William R. Robinson (Carolina Power & Light Company) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Response to Generic Letter 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,"* HNP 96-186, October 30, 1996.

VII-12 Letter from William R. Robinson (Carolina Power & Light Company) to USNRC Document Control Desk, *Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 NRC Generic Letter 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,"120-dayResponse,"* HNP 97-011, January 28, 1997.

VII-13 Letter from Lisa M. Regner (USNRC) to Robert J. Duncan II (Carolina Power & Light Company), *Shearon Harris Nuclear Power Plant, Unit 1- Response to Generic Letter 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,* TAC No. M96818, August 10, 2007.

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VIII. CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

1. A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate

   A. a description of the change

   B. identification of analyses affected by and/or supporting the change

   C. justification for the change, including the type of information discussed in Section III above, for any analyses that support and/or are affected by change

RESPONSE TO VIII. CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

VIII.1 Technical Specification Changes

The proposed TS changes are described in detail in Enclosure 1, Section 2.

VIII.2 Protection System Settings Changes

The proposed protection system setpoint changes are described in detail in Enclosure 1, Section 2.

VIII.3 Emergency System Settings Changes

There are no emergency system setpoint changes resulting from the MUR power uprate, although some instruments will require rescaling to support implementation.

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REGULATORY COMMITMENTS

The actions in this document committed to by Harris Nuclear Plant (HNP) are identified in the following table. Statements in this submittal, with the exception of those in the table below, are provided for information purposes and are not considered regulatory commitments. Please direct any questions regarding this document or any associated regulatory commitments to the HNP Supervisor - Licensing/Regulatory Programs.

Item	Commitment	Completion Date
1	Relocated Technical Specifications and Design Basis Requirements procedure (PLP-114) will be revised to include LEFM controls (Enclosure 1 Section 2.5, Enclosure 2 Section I.1.H)	Prior to operating above 2900 MWt (approximately 98.4% RTP)
2	Procedures and documents for the new LEFM will be established or revised (Enclosure 2 Sections I.1.D.i.a, I.1.H, VII.2.A).	Prior to operating above 2900 MWt (approximately 98.4% RTP)
3	Appropriate personnel will receive training on the LEFM and affected procedures (Enclosure 2 Sections I.1.D.i.a, VII.2.A, and VII.2.D).	Prior to operating above 2900 MWt (approximately 98.4% RTP)
4	Plant electrical output will be limited to the capability of the existing main transformers. (Enclosure 2 Section V.1.F.iii).	Prior to installation of replacement main transformers.
5	Simulator changes and validation will be completed (Enclosure 2 Section VII.2.C).	Prior to operating above 2900 MWt (approximately 98.4% RTP)
6	Existing plant operating procedures related to temporary operation above full steady-state licensed power levels will be revised, as necessary (Enclosure 2 Section VII.4).	Prior to operating above 2900 MWt (approximately 98.4% RTP)

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Item	Commitment	Completion Date
7	Required plant modifications will be completed (Enclosure 2 Section VII.3)	Prior to operating above 2900 MWt (approximately 98.4% RTP)
8	Plant electrical output will be limited to the capacity of the existing isolated phase bus. (Enclosure 2 Section V.1.F.ii).	Prior to upgrading the isolated phase bus.

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CAMERON
AFFIDAVIT PURSUANT TO 10 CFR 2.390
FOR WITHHOLDING PROPRIETARY INFORMATION
(7 Pages)

Measurement Systems

Caldon® Ultrasonics Technology Center
1000 McClaren Woods Drive
Coraopolis, PA 15108
Tel 724-273-9300
Fax 724-273-9301
www.c-a-m.com

CAMERON

January 18, 2011
CAW 11-01

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject:

1. Caldon® Ultrasonics Engineering Report ER-697 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at Harris Unit 1 Using the LEFM✓ + System"
2. Caldon® Ultrasonics Engineering Report ER-720 Rev. 2, "Meter Factor Calculation and Accuracy Assessment for Harris Nuclear Plant"

Gentlemen:

This application for withholding is submitted by Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 11-01 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 11-01 and should be addressed to the undersigned.

Very truly yours,

Cal Hastings

Cal Hastings
General Manager

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

January 18, 2011
CAW 11-01

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Cal Hastings, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

C. Hastings
Cal Hastings
General Manager

Sworn to and subscribed before me

this 18th day of

January, 2011

Joann B. Thomas
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2011
Member, Pennsylvania Association of Notaries

1. I am the General Manager of Caldon Ultrasonics Technology Center, and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Cameron.

2. I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information. The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

4. Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Cameron.

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Cameron system, which include the following:

(a) The use of such information by Cameron gives Cameron a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Cameron competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Cameron ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Cameron at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Cameron of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Cameron in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Cameron capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10 CFR §§ 2. 390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld are the submittals titled:

- Caldon® Ultrasonics Engineering Report ER-697 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at Harris Unit 1 Using the LEFM✓+ System "
- Caldon® Ultrasonics Engineering Report ER-720 Rev. 2, "Meter Factor Calculation and Accuracy Assessment for Harris Nuclear Plant."

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for LEFM CheckPlus System used by Harris Unit 1 for an MUR UPRATE.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Cameron effort and the expenditure of a considerable sum of money.

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, similar technical programs would have to be performed, and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.